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10 CFR 50.90

February 15, 2019

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

R. E. Ginna Nuclear Power Plant
Renewed Facility Operating License No. DPR-18
NRC Docket No. 50-244

Subject: License Amendment Request (LAR) to Revise Technical Specification (TS) 5.5.15, "Containment Leakage Rate Testing Program," for Permanent Extension of Type A Leak Rate Test Frequencies

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to the Technical Specification (TS) 5.5.15, "Containment Leakage Rate Testing Program," for R. E. Ginna Nuclear Power Plant (Ginna).

The proposed change revises TS 5.5.15 to reflect an increase to the existing Type A Integrated Leak Rate Test (ILRT) program test interval from 10 years to 15 years in accordance with Nuclear Energy Institute (NEI) Report NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2-A. The proposed change will also reflect adoption of both the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, "Containment System Leakage Testing Requirements," and a more conservative allowable test interval extension of nine months for Type A leakage tests in accordance with NEI 94-01, Revision 2-A. This LAR also proposes an administrative change to remove the exception under TS 5.5.15 for the one-time 15-year Type A test internal being performed after May 31, 1996, and performed prior to May 31, 2011, as this has already occurred.

The proposed change has been reviewed by the Ginna Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed amendment by February 15, 2020, to support the extension of the ILRT, which is required to be performed during the refueling outage scheduled in the spring of 2020. Once approved, the amendment will be implemented within 30 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of New York of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Jessie Hodge at (610) 765-5532.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of February 2019.

Respectfully,

David T. Gudger for

James Barstow
Director - Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1. Evaluation of Proposed Change
2. Markup of Technical Specifications Page
3. Retyped Version of Technical Specifications Page
4. Ginna Evaluation of Risk Significance of Permanent ILRT Extension

cc:	USNRC Region I, Regional Administrator	w/ attachments
	USNRC Senior Resident Inspector, Ginna	"
	USNRC Project Manager, Ginna	"
	A. L. Peterson, NYSERDA	"

ATTACHMENT 1

Evaluation of Proposed Changes

R. E. Ginna Nuclear Power Plant

Renewed Facility Operating License No. DPR-18

Docket No. 50-244

Subject: **License Amendment Request (LAR) to Revise Technical Specification (TS) 5.5.15, "Containment Leakage Rate Testing Program," for Permanent Extension of Type A Leak Rate Test Frequencies**

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, “Application for amendment of license, construction permit or early site permit,“ Exelon Generation Corporation, LLC (EGC) requests an amendment to Renewed Facility Operating License DPR-18 for the R. E. Ginna Nuclear Power Plant (GNPP) to allow for permanent extension of the Type A testing frequency. The proposed change revises Technical Specification (TS) 5.5.15, “Containment Leakage Rate Testing Program,” to reflect the following:

- Increase the existing Type A Integrated Leak Rate Test (ILRT) program test interval from 10 years to 15 years in accordance with Nuclear Energy Institute (NEI) Report NEI 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” Revision 2-A (Reference 1).

- Adopt the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, “Containment System Leakage Testing Requirements” (Reference 2).

- Adopt a more conservative allowable test interval extension of nine months, for Type A leakage tests in accordance with NEI 94-01, Revision 2-A (Reference 1).

Specifically, the proposed change contained herein revises GNPP TS 5.5.15 by replacing the references to Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program” (Reference 3), and 10 CFR 50, Appendix J, Option B, with a reference to NEI 94-01, Revision 2-A (Reference 1) as the document used by EGC for GNPP to implement a performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

This LAR also proposes an administrative change to remove the exception under TS 5.5.15 for the one-time 15-year Type A test interval being performed after May 31, 1996, and performed prior to May 31, 2011, as this has already occurred.

2.0 DETAILED DESCRIPTION

GNPP TS 5.5.15, “Containment Leakage Rate Testing Program,” currently states, in part:

> “A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J”:
>
> a. Section 9.2.3: The first Type A test performed after the May 31, 1996 Type A test shall be performed by May 31, 2011.”

The proposed changes to the GNPP TS 5.5.15 will replace the reference to RG 1.163 with reference to NEI Topical Report NEI 94-01, Revision 2-A. This LAR also proposes to remove the exception under TS 5.5.15, paragraph a. for the first Type A test performed after May 31, 1996, being performed by May 31, 2011, as this test has already occurred, and thereby is no longer applicable.

The proposed change revises the GNPP TS 5.5.15 to state, in part (with recommended changes using strike-out and **bold-type** for clarification purposes):

"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 exception to~~ NEI 94-01, ~~Rev. 0,~~ "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," **Revision 2-A, dated October 2008**.

~~a. Section 9.2.3: The first Type A test performed after the May 31, 1996 Type A test shall be performed by May 31, 2011."~~

Therefore, the retyped ("clean") version of TS 5.5.15 will appear as follows:

"A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, Industry Guideline for implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 2-A, dated October 2008."

The marked-up pages for GNPP TS 5.5.15 are provided in Attachment 2.

Attachment 4 contains the plant specific risk assessment conducted to support this proposed change. This risk assessment followed the guidelines of RG 1.174, Revision 3 (Reference 4) and RG 1.200, Revision 2 (Reference 5). The risk assessment concluded that increasing the ILRT on a permanent basis to a one-in-fifteen-year frequency is considered to represent a small change in the GNPP risk profile.

3.0 TECHNICAL ANALYSIS

3.1 Description of Containment System

The reactor Containment Structure is a reinforced concrete, vertical right cylinder with a flat base and a hemispherical dome. It ensures that leakage of radioactive materials to the environment is minimized even if gross failure of the Reactor Coolant System (RCS) were to occur. A welded steel liner is attached to the inside face of the concrete shell to ensure a high degree of leak tightness. The thickness of the liner in the cylinder and dome is 3/8 inch (in.) and in the base it is 1/4 in. The cylindrical reinforced concrete walls are 3 feet (ft.) 6 in. thick, and the concrete hemispherical dome is 2 ft. 6 in. thick. The concrete base slab is 2 ft. thick with an additional 2-ft.-thick concrete fill over the bottom liner plate. The Containment Structure is 99 ft. high to the spring line of the dome and has an inside diameter of 105 ft. The containment vessel provides a minimum free volume of approximately 972,000 cubic feet (ft^3). The reactor vessel is located in the center of the Containment Structure below ground level.

The Containment Structure at GNPP employs the use of post-tensioned pre-stressing tendons. There are 160 vertical tendons in the cylindrical walls of containment and no tendons in the dome. A tendon is comprised of 90 post-tensioned wires, which run through a sheath (e.g., pipe) in the containment wall. The tendon is anchored at both the top and bottom of the cylindrical wall; however, the bottom is connected to rock anchors such that

this anchorage is typically inaccessible. The tendon can is filled with grease as a corrosion preventative measure. Located within the tendon can are the button-heads, anchor head and shims. The button-heads are the cold-formed ends of the 90 wires, which anchor them to the anchor head. The shims maintain the space between the anchor head and bearing plate once the tendon is stressed, keeping force on the building.

The containment leakage pressure boundary is provided by the single steel liner in the containment vessel. Each system whose piping penetrates this boundary is designed to maintain isolation of the containment from the outside environment. The Containment Structure and all penetrations are designed to withstand, within design limits, the combined loadings of the Design-Basis Accident (DBA) and design seismic conditions. All piping systems, which penetrate the containment, are anchored in the penetration sleeve or the structural concrete of the Containment Structure. The penetrations for the main steam, feedwater, blowdown, and sample lines are designed so that the penetration is stronger than the piping system and the containment will not be breached due to a postulated pipe rupture. The liner thickness in the vicinity of typical penetrations is increased to a minimum of 3/4 in. All lines connected to the primary coolant system that penetrate the containment are also anchored in the secondary shield walls (i.e., walls surrounding the steam generators and reactor coolant pumps) and are each provided with at least one valve between the anchor and the RCS. For mechanical penetrations that interface with hot fluid systems, a containment penetration cooling system is used to prevent the bulk concrete temperature surrounding the penetrations from exceeding 150°F (degrees Fahrenheit). Containment electrical penetrations are designed so the Containment Structure can, without exceeding the design leakage rate, accommodate the postulated environment resulting from a loss-of-coolant accident. The electrical penetrations have been shown to maintain structural integrity when subjected to mechanical stresses caused by large magnitude fault currents.

3.1.1 Fuel Transfer Penetration

A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pool. The penetration consists of a stainless steel pipe installed inside a larger pipe. The inner pipe acts as the transfer tube and connects the refueling canal with the spent fuel pool. The tube is fitted with a standard stainless steel flange in the refueling canal and a stainless steel sluice gate valve in the spent fuel pool. The outer pipe is welded to the containment liner. The fuel transfer penetration, like all other penetrations, is anchored in the containment shell. Because this anchor point moves when the containment vessel is subjected to load, expansion joints are provided where the penetration is connected to structures inside and outside of the containment vessel. Since the penetration is located on a skewed angle, not normal to the containment shell, the expansion joints are subjected to both radial and tangential (lateral) motions. The expansion bellows inside the containment vessel provide a water seal for the refueling canal and accommodate thermal growth of the penetration from the anchor, as well as the pressure and earthquake produced motion of the anchor (the containment shell). The expansion joint accommodates motion of the sleeve within the containment shell relative to the portion of the sleeve anchored in the wall of the refueling canal in the Auxiliary Building.

3.1.2 Equipment Hatch and Personnel Hatch

An equipment hatch, constructed of welded steel and having a double-gasketed flange and bolted dished door, is located near grade. The equipment hatch has a diameter of 14 ft. and is used for transportation of equipment through the containment wall.

Two personnel accesses are provided. One personnel hatch penetrates the dished door of the equipment hatch. The other is located diametrically opposite the equipment hatch. Each personnel hatch is a hydraulically-latched double door, welded steel assembly. An equalizing valve connects each personnel hatch with the interior of the containment vessel for the purpose of equalizing pressure in the personnel hatch with that in the containment. Hatch closures are the double-tongue, single gasket type. The access locks are properly interlocked to ensure door closure at all times.

3.2 Emergency Core Cooling System Net Positive Suction Head Analysis

The Emergency Core Cooling System (ECCS) is designed so that adequate Net Positive Suction Head (NPSH) is provided to system pumps. In addition to considering the static head and suction line pressure drop, the calculation of available NPSH in the recirculation mode for the Residual Heat Removal (RHR) pumps assumes that the vapor pressure of the liquid in the sump equals containment pressure. GNPP analysis does not rely on NPSH above 0 pounds per square inch gauge (psig) pressure and does not credit accident pressure (containment over-pressurization) to provide adequate NPSH for ECCS pumps.

The NPSH of the RHR pumps is evaluated for normal plant shutdown operation and for both the injection and recirculation phase operations of the DBA. Recirculation operation gives the limiting NPSH requirements and the NPSH available is determined from the containment water level, the temperature and pressure of the sump water, and the pressure drop in the suction piping from the sump to the pumps. Adequate margin between required and available NPSH exists under all required operating conditions.

The NPSH for the Safety Injection (SI) pumps is evaluated for both the injection and recirculation phase operations of the DBA. The end of injection phase operation gives the limiting NPSH requirement and the NPSH available is determined from the elevation head and vapor pressure of the water in the Refueling Water Storage Tank (RWST), and the pressure drop in the suction piping from the tank to the pumps.

The NPSH for the Containment Spray (CS) pump is evaluated for both the injection and recirculation phase operations of the DBA. The end of the injection phase operation gives the limiting NPSH requirement and the NPSH available is determined from the elevation head and vapor pressure of the water in the RWST, and the pressure drop in the suction piping from the tank to the pumps.

The NPSH evaluation was based on a hydraulic analysis model that is set to maximize flow from the operating pumps and minimize pump suction pressure. Several cases were run to determine the limiting case for NPSH. That is: two (2) RHR Pumps, three (3) (50%) SI Pumps and two (2) CS Pumps. There are three (3) Charging Pumps, but these are not safety related and are not credited post-accident in the Updated Final Safety Analysis Report (UFSAR), Chapter 15 analysis. The NPSH evaluation is based on a hydraulic flow evaluation with all pumps initially operating at the maximum design flow rates and then

limits the operating pumps based on procedural direction contained in the Emergency Operating Procedure (EOP). The direction in this procedure was setup to improve NPSH margin and maintain adequate core cooling. This ensures that adequate margin between required and available NPSH exists under all postulated operating conditions. The actual available NPSH is always greater than the calculated NPSH. The CS pumps are only required to operate during the injection phase for DBAs. Analyses have been performed to allow CS pump operation in beyond design basis events and maintain adequate NPSH, but these cases do credit containment overpressure.

3.3 Justification for the TS Change

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

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in the TS. 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 DBA. Appendix J identifies three types of required tests:

1) Type A tests, intended to measure the primary containment overall integrated leakage rate;

2) Type B tests, intended to detect local leaks and to measure leakage across pressure-containing or leakage limiting boundaries (other than valves) for primary containment penetrations; and,

3) Type C tests, intended to measure Containment Isolation Valve (CIV) leakage rates. 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 Types 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, RG 1.163 (Reference 3) was issued. The RG endorsed NEI 94-01, Revision 0, (Reference 6) with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A

tests) to reduce the test frequency for the containment Type A (ILRT) test from three (3) tests in ten (10) years to one (1) test in ten (10) years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, (Reference 7) and Electric Power Research Institute (EPRI) TR-104285 (Reference 8), both of which showed that the risk increase associated with extending the ILRT surveillance interval was very small. In addition to the 10-year ILRT interval, provisions for extending the test interval an additional 15 months were considered in the establishment of the intervals allowed by RG 1.163 and NEI 94-01, but that this extension of interval "should be used only in cases where refueling schedules have been changed to accommodate other factors."

In 2008, NEI 94-01, Revision 2-A (Reference 1), 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 NRC Safety Evaluation Report (SER) on NEI 94-01. NEI 94-01, Revision 2-A, includes provisions for extending Type A ILRT intervals to up to 15 years and incorporates the regulatory positions stated in RG 1.163 (Reference 3). 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.

The NRC has provided guidance concerning the use of test interval extensions in the deferral of ILRTs beyond the 15-year interval in NEI 94-01, Revision 2-A, NRC SER Section 3.1.1.2 that 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.

In 2012, NEI 94-01, Revision 3-A (Reference 9), was issued. This document describes an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J and includes provisions for extending Type A ILRT intervals to up to 15 years. RG 1.163 and NRC SERs dated June 25, 2008, and June 8, 2012 (References 3, 10 and 11, respectively) endorse NEI 94-01 as an acceptable methodology for complying with the provisions of 10 CFR Part 50, Appendix J, Option B. The regulatory positions stated in RG 1.163, as modified by References 7 and 8, are incorporated in NEI 94-01 Rev. 3-A. 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. Extensions of Type B and Type C test intervals are allowed based upon completion of two consecutive periodic As-Found (AF) 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 AF tests, schedule and review as described in NEI 94-01, Revision 3-A, Section 11.3.2.

GNPP has evaluated the additional extension of Type C intervals afforded by NEI 94-01, Revision 3-A, and has chosen not to adopt NEI 94-01, Revision 3-A at this time. However, the risk assessment performed to permanently extend the currently allowed containment Type A ILRT to fifteen years used the methodology currently endorsed by NEI 94-01, Revision 3-A for the required confirmatory risk impact assessment, as this is the most up to date guidance available.

3.3.2 Current GNPP Primary Containment Leakage Rate Testing Program Requirements

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to choose containment leakage testing under either Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." On February 13, 1996, the NRC approved TS Amendment 61 for GNPP (Reference 12) authorizing the implementation of 10 CFR 50, Appendix J, Option B for Types A, B and C tests.

Currently, TS 5.5.15 requires that a program be established to comply with the containment leakage rate testing requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B. The program is required to be in accordance with the guidelines contained in RG 1.163. This RG endorses, with certain exceptions, NEI 94--01, Revision 0 (Reference 6), as an acceptable method for complying with the provisions of Appendix J, Option B.

RG 1.163, Section C.1 states that licensees intending to comply with 10 CFR 50, Appendix J, Option B, should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 rather than using test intervals specified in ANSI/ANS 56.8-1994 (Reference 13). NEI 94-01, Section 11.0 refers to Section 9, which states that Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per ten years based on acceptable performance history. Acceptable performance history is defined as completion of two (2) consecutive periodic Type A tests where the calculated performance leakage was less than $1.0L_a$ (where L_a is the maximum allowable leakage rate at the calculated peak post-accident pressure). Elapsed time between the first and last tests in a series of consecutive satisfactory tests used to determine performance shall be at least 24 months.

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" (AF) leakage history to determine a frequency for leakage testing which provides assurance that leakage limits will not be exceeded. 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 (Reference 7). The evaluation documented in NUREG-1493 includes a study of the dependence or reactor

accident risks on containment leak tightness. NUREG-1493 concludes in Section 10.1.2 that reducing the frequency of Type A tests (ILRT) from the original three (3) tests per 10 years to one (1) 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.

3.3.3 GNPP 10 CFR 50, Appendix J, Option B Licensing History

February 13, 1996

The NRC issued Amendment No. 61, which modified TS 5.5.15 to allow the use of 10 CFR Part 50, Appendix J, Option B, Performance-Based Containment Leakage Rate Testing. (Reference12)

December 8, 2005

The NRC issued Amendment No. 93, which revised TS 5.5.15 to extend on a one-time basis, the interval for completing the next Type A test, pursuant to 10 CFR 50, Appendix J, from 10 years to 15 years. This amendment added the following exception wording to TS 5.5.15: “The first test performed after the May 31, 1996 test shall be performed by May 31, 2011.” (Reference 14)

3.3.4 Integrated Leakage Rate Testing (ILRT) History

As noted previously, GNPP TS 5.5.15 currently requires Types A, B, and C testing to be performed in accordance with RG 1.163, which endorses the methodology for complying with Option B. Since the adoption of Option B, the performance leakage rates are calculated in accordance with NEI 94-01, Revision 0, Section 9.1.1 for Type A testing. Table 3.3.4-1 lists the past Periodic Type A ILRT results for GNPP.

Table 3.3.4-1 – GNPP Type A Testing History					
Test Date	95% Upper Confidence Limit (wt.%/day)	As-Found Leakage (wt.%/day)	Acceptance Criteria (wt.%/day)	As-Left Leakage (wt.%/day)	Acceptance Criteria (0.75La) (wt.%/day)
11/1969	N/A	(1)	(1)	0.0387	0.0731
(2) 10/1972	N/A	0.07838	0.1146	0.0620	0.1146
(2) 2/1976	N/A	(3)	0.1146	0.0440	0.1146
(2) 3/1978	0.04900	0.05092	0.1146	0.0490	0.1146
(2) 3/1982	0.01970	(4)	0.1146	0.0197	0.1146
(2) 3/1986	0.06407	0.06741	0.1146	0.06407	0.1146
(2) 5/1989	0.04631	0.04632	0.1146	0.04631	0.1146
(2) 4/15/1993	0.05383	0.05554	0.1146	0.05550	0.1146
6/10/1996	0.11967	0.11969	(5) & (6) 0.2000	0.11967	(5) & (6) 0.1500
6/2/2011	0.10260	0.1313	0.2000	0.1078	0.1500

Table 3.3.4-1 Notes:

Note 1: This ILRT was a Pre-Operational Test; therefore, no As-Found data exists.

Note 2: Reduced pressure test performed at 35 psig. The ILRT completed in 11/1969 was performed at full pressure and reduced pressure conditions. A full pressure test was performed at 60 psig and a reduced pressure test at 35 psig. Subsequent tests after the pre-operational test, in 1969, were conducted at reduced pressure (35 psig), until 1994, when ANSI/ANS 56.8 required ILRT test pressure to be greater than (0.96) times (peak post-accident pressure (Pa)).

Note 3: The ILRT that began on February 7, 1976, was aborted after approximately seven hours into the equalization period due to leakage that exceeded the test acceptance criteria of 0.75La (0.1146 wt.%/day).

The failure of this Type A test was the result of leakage through the purge exhaust dampers and check valve 1713. Abnormally high leakage rates through the purge dampers have historically occurred only when the plant is in the cold shutdown condition. The pathway from containment through check valve 1713 leads into a normally closed nitrogen system outside containment, which is pressurized to 90 psig. Venting of this system both inside and outside containment was performed especially for the Type A test.

In September 1987, the containment purge system at penetration 204 (purge air supply) and penetration 300 (purge air exhaust) were modified to provide greater assurance of containment integrity. The 48 in. inboard CIV at each of these two penetrations was removed and replaced with a blind flange. The blind flanges on penetrations 204 and 300 are removed and reassembled during plant outages, when required. These blind flanges remain in place when the reactor is in a mode where containment integrity is required.

Note 4: The GNPP 1982 As-Found Leakage rate exceeded its limit of 0.75La (0.1146 wt.%/day). After plant shutdown for the 1982 refueling outage (RFO), AF Types B and C tests were performed. During this testing, the leakage rates for the

Purge Supply (5870) and Exhaust (5878), as well as check valve 1599, were found to be excessive. The measured leak rates on the Purge Supply (5870) and Purge Exhaust (5878) were 15,593 cubic centimeters per minute (cc/min) and 244,142 cc/min, respectively. The leak rate on check valve 1599 was not quantified. Because of this, the Leakage Savings for this ILRT could not be determined and the AF leak rate could not be calculated. This ILRT was performed in March 1982. It should be noted that the Type C total leak rate summation at the end of 1981 was 1,016 cc/min. The measured leak rates for the Purge Supply (5870) and Exhaust (5878) were found to be 174 cc/min and 362 cc/min, respectively, on November 18, 1981.

Check valve 1599 was subsequently replaced in November 1982 with an Air-Operated Valve (AOV) to provide greater assurance of proper valve operability and leak tight closure.

In September 1987, the containment purge system at penetration 204 (purge air supply) and penetration 300 (purge air exhaust) were modified to provide greater assurance of containment integrity. The 48 in. inboard CIVs at each of these two penetrations were removed and replaced with a blind flange. The blind flanges on penetrations 204 and 300 are removed and reassembled during plant outages, when required. These blind flanges remain in place when the reactor is in a mode where containment integrity is required.

Note 5: NEI 94-01, Revision 0 issued on July 26, 1995, specifies the acceptance criteria of 1.0 La for the AF leakage ILRT leakage. Prior to this, the acceptance criterion for the AF ILRT leakage was 0.75 L_a. Therefore, for ILRT tests performed after July 26, 1995, at GNPP, the acceptance criteria specified as wt.%/day was increased from 0.15 wt.%/day to 0.2 wt.%/day. This is assuming a test pressure of Pa. See Note 4 regarding testing at reduced pressures where test pressure is less than Pa.

Note 6: ANSI/ANS 56.8 (1994) was approved on August 4, 1994. In Section 3.2.11, it states “The Type A test pressure shall not be less than 0.96Pa.” This eliminated the option to perform Type A testing at a reduced pressure. ILRT tests performed subsequent to ANSI/ANS 56.8 (1994) were tested at Pa (60 psig). Therefore, tests performed after 1994 did not require a partial pressure calculation to adjust the wt.%/day for the reduced test pressure. See Note 2 above.

3.3.5 Performance Leakage Rate Determination

NEI 94-01 defines the performance leakage rate, or performance criteria, for the Type A ILRT as allowable leakage less than $1.0L_a$. Extensions of the Type A ILRT are allowed based upon two consecutive, periodic Type A ILRTs where the performance leakage rate is less than $1.0L_a$. The past two ILRTs for GNPP (1996 and 2011) both had measured performance leakage rates less than $1.0L_a$ (0.2 wt.%/day). Since both tests were satisfactory, the GNPP ILRT remains on an extended frequency. The current ILRT frequency for GNPP is 10 years. Table 3.3.4-2 provides a breakdown of the calculation of the performance leakage rate for the past two ILRTs at GNPP.

Table 3.3.4.2 – Verification of Current Extended ILRT Interval for GNPP

Test Date	95% UCL Leakage Rate (wt.%/day)	Water Level Corrections (wt.%/day)	Corrections for Valves Isolated during Test (wt.%/day)	Types B and C Penalties Due to Excessive Leakage (wt.%/day)	Performance Leak Rate (Acceptance Criteria ≤ 0.5 (wt.%/day)	Test Method
6/10/1996	0.11967	0.0000	0.00002	0.0000	0.11969	Mass Pt.
6/2/2011	0.1026	0.0032	0.0020	0.0000	0.10780	Mass Pt.

3.4 Plant Specific Confirmatory Analysis

3.4.1 Methodology

A plant specific confirmatory analysis was performed to provide a risk assessment of extending the currently allowed containment Type A ILRT to a permanent interval of fifteen years. The risk assessment follows the guidelines from:

1. NEI 94-01, Revision 3-A (Reference 9),
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 16),
3. The NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) as stated in RG 1.200 (Reference 5) as applied to ILRT interval extensions.
4. Risk insights in support of a request for a plant's licensing basis as outlined in RG 1.174 (Reference 4).
5. 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 17),
6. The methodology used in EPRI 1018243, Revision 2-A of EPRI 1009325 (Reference 15).

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 (15 years in this extension analysis). 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 1.0L_a.

The basis for the current 10-year test interval was established in 1995 during development of the performance-based Option B to Appendix J and is provided in Section 11.0 of NEI 94-01, Revision 0. Section 11.0 of NEI 94-01 (Reference 6) states that NUREG-1493, "Performance-Based Containment Leak Test Program," 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 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 pressurized water reactor (PWR) plant (i.e., Surry), 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 GNPP.

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 15). 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 changes for a 15-year interval.

It should be noted that containment leak-tight integrity is also verified through periodic inservice inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI. More specifically, Subsection IWE provides the rules and requirements for inservice inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. 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.

In the SER issued by the NRC letter dated June 25, 2008 (Reference 10), the NRC concluded that the methodology in EPRI TR-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 (SE). Table 3.4.1-1 addresses each of the four limitations and conditions for the use of EPRI 1009325, Revision 2.

Table 3.4.1-1 – EPRI TR-1009325 Revision 2 Limitations and Conditions

Limitation/Condition (From Section 4.2 of SER)	GNPP Response
1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension.	GNPP PRA technical adequacy is addressed in Section 3.4.2 of this LAR and Attachment 2, "GNPP Evaluation of Risk Significance of Permanent ILRT Extension," Appendix A, PRA Model Technical Adequacy.
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 SER.	RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis . RG 1.174 defines very small changes in risk as resulting in increases of Core Damage Frequency (CDF) less than 1.0E-06/year and increases in Large Early Release Frequency (LERF) less than 1.0E-07/year. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated as 9.52E-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. Therefore, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of RG 1.174 (Reference 4). The risk change resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years bounds the 1 in 10 years to 1 in 15 years risk change. Considering the increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 10 years to 1 in 15 years is estimated as 3.96E-8, the risk increase is "very small" using the acceptance guidelines of RG 1.174.

Table 3.4.1-1 – EPRI TR-1009325 Revision 2 Limitations and Conditions	
Limitation/Condition (From Section 4.2 of SER)	**GNPP Response**
2.a (continued)	When external event risk is included, the increase in 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 3.91E-7/year using the EPRI guidance, and total LERF is 1.61E-6/year. As such, the estimated change in LERF is determined to be "small" using the acceptance guidelines of RG 1.174 (Reference 4). The risk change resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years bounds the 1 in 10 years to 1 in 15 years risk change. When external event risk is included, the increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 10 years to 1 in 15 years is estimated as 2.28E-7 and the total LERF is 1.45E-6. Therefore, the risk increase is "small" using the acceptance guidelines of RG 1.174 (Reference 4). As discussed in Sections 5.1.3 and 5.2.7 of Attachment 2 of this submittal, the EPRI methodology used to estimate the increase in LERF and the models used to estimate total LERF are conservative. Therefore, the conservative methodology adds margin.
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 1-per-15 years, measured as an increase to the total integrated plant risk for those accident sequences influenced by Type A testing, is 0.29 person-rem/year. NEI 94-01 states that a 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.

Table 3.4.1-1 – EPRI TR-1009325 Revision 2 Limitations and Conditions	
Limitation/Condition (From Section 4.2 of SER)	**GNPP Response**
2.c In addition, a small increase in Conditional Containment Failure Probability (CCFP) 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 3 in 10 year interval to 1 in 15 year interval is 0.881%. NEI 94-01 states increases in CCFP of ≤ 1.5% are small. Therefore, this increase is judged to be small.
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 accident case (accident case 3b) used by the licensees shall be 100 La instead of 35 La.	The representative containment leakage for Class 3b sequences is 100La based on the guidance provided in EPRI Report No. 1009325, Revision 2-A (EPRI 1018243). It should be noted that this is more conservative than the earlier previous industry ILRT extension requests, which utilized 35 La for the Class 3b sequences.
4. A licensee amendment request (LAR) is required in instances where containment over-pressure is relied upon for ECCS performance.	Containment overpressure is not required for ECCS Performance and is discussed in Section 3.2 of this enclosure. Therefore, no additional request is necessary.

3.4.2 Technical Adequacy of the GNPP Probabilistic Risk Assessment

3.4.2.1 PRA Quality Statement for Permanent 15 Year ILRT Extension

The GN116A version of the GNPP PRA model is the most recent evaluation of internal event risks. The GNPP PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the GNPP PRA is based on the event tree/fault tree methodology, which is a well-known methodology in the industry.

EGC employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process and the use of self-assessments and independent peer reviews. Prior to joining the Exelon nuclear fleet in 2014, comparable practices were in place when GNPP was owned and operated by Constellation Energy Nuclear Group (CENG). Because of the similarities between the CENG and Exelon practices, no additional discussion specifically regarding the legacy CENG approach will be provided. The following information describes the Exelon approach (and by extension, the CENG approach) to PRA model maintenance, as it applies to the GNPP PRA.

3.4.2.2 PRA Maintenance and Update

The EGC risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plants. This process is defined in the Exelon Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation training and reference materials (T&RMs).

- Exelon T&RM ER-AA-600-1015, "Full Power Internal Event (FPIE) PRA Model Update," delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Exelon nuclear generation sites.
- ER-AA-600-1061, "Fire PRA Model Update and Control" delineates the responsibilities and guidelines for updating the station fire PRA.

The overall Exelon Risk Management program, including ER-AA-600-1015 and ER-AA-6001061, defines the process for implementing regularly scheduled and interim PRA model updates, tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.), and controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for Exelon nuclear generation sites.
- As an NFPA 805 plant, all modifications are reviewed to ensure the modification meets the fire requirements during the initial design phase of a modification per ER-AA-600-1068 .
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on Systems, Structures, and Components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65(a)(4)).

As indicated previously, RG 1.200 also requires that additional information be provided as part of the LAR submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated in to the PRA model, relevant peer review findings, consistency with applicable PRA Standards, and the identification of key assumptions) will be discussed in turn.

3.4.2.3 Plant Changes Not Yet Incorporated into the PRA Model

Each Exelon station maintains an Updating Requirements Evaluation (URE) database to track all enhancements, corrections, and unincorporated plant

changes. During the normal screening conducted as part of the plant change process, if a potential model update is identified, a new URE database item is created. Depending on the potential impact of the identified change, the requirements for incorporation will vary.

As part of this PRA evaluation, a review of open items in the URE database for GNPP is performed, and an assessment of the impact on the results of the application is made. Some UREs may affect the LERF containment modeling. Open URE 834 pertains to finding and observation (F&O) LE-C10-01, which states credit for scrubbing was not taken. Per the response to request for additional information (RAI) 17 for the TS Task Force (TSTF)-425 LAR (Reference 18), scrubbing may be applicable to the following three containment bypass conditions: 1) a steam generator tube rupture event with feedwater available, or 2) internal flood scenarios with an interfacing system Loss Of Coolant Accident (LOCA) and the affected auxiliary building room flooded, or 3) sequences where the Interfacing System LOCA (ISLOCA) break is in the RHR pits. Since these are only Class 8 (SGTR or ISLOCA) sequences, there would be no effect on ILRT ΔLERF (change in Class 3b risk).

Open URE 837 pertains to LERF quantification, which is used in this analysis, and tracks finding LE-C9a, which is Capability Category (CC) I. Since NEI 94-01 endorses using PRA models conformed to CC- I of the ASME/ANS standard, the GNPP PRA model is of sufficient quality to use for this ILRT analysis. Additionally, this IRLT extension analysis has significant margin to the Regulatory Guide 1.174 acceptance guidelines (Reference 4), so any model update from this URE is judged to be sufficiently small so as to not affect this IRLT extension analysis.

Additional open UREs may also affect overall CDF and LERF quantification results, which are used to calculate change in risk metrics for the ILRT extension evaluation. After evaluating all open UREs for their effect on CDF and LERF, it is concluded the aggregate of open UREs leads to the current PRA model being conservative. A conservative CDF alone would lead to conservative calculations for the ILRT extension; since LERF is subtracted from CDF to calculate the risk increase due to the ILRT extension, if the LERF conservatism is greater than the CDF conservatism, the ILRT extension calculations would not be conservative. Since the magnitude of CDF and LERF conservatisms are unknown, a sensitivity is performed where LERF is not subtracted from CDF when calculating the change in risk for the ILRT extension (as described in Section 5.2.1 in Appendix B of Attachment 2 of this submittal). Sensitivity results are shown in Appendix B of Attachment 2 of this submittal.
The increase in the overall probability of LERF due to Class 3b sequences is less than 10-7. Therefore, the ΔLERF is considered very small (Reference 4).

NEI 94-01 (Reference 9) states that a 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 in Appendix B of Attachment 2 of this submittal, the results of this calculation meet the dose rate criteria.

As stated in Section 2.0 in Appendix B of Attachment 2 of this submittal, 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 0.917%. Therefore, this increase is judged to be small.

This sensitivity methodology is also applied to the external event risk ILRT extension calculations.

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 due to increasing the ILRT interval from 3 to 15 years is 4.03E-7, which meets the guidance for small change in risk, as it exceeds 1.0E-7/yr and remains less than a 1.0E-6 change in LERF. For this change in LERF to be acceptable, total LERF must be less than 1.0E-5.

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.

3.4.2.4 Applicability of Peer Review Findings and Observations (F&Os)

The peer review process demonstrates the technical acceptability of the GNPP PRA models. The purpose of the industry PRA peer review process is to provide a method for establishing the technical capability and adequacy of a PRA relative to expectations of knowledgeable practitioners, using a set of guidance that establishes a set of minimum requirements. PRA peer reviews continue to be performed as PRAs are updated (and upgraded) to ensure the ability to support risk-informed applications and has proven to be a valuable process for establishing technical adequacy of nuclear power plant PRAs. There have been three relevant peer reviews conducted on the current PRA model:

- The 2009 peer review for the PRA ASME model update identified 309 Supporting Requirements (SRs) applicable to the GNPP PRA. Of these 29 were not met, 2 met CC 1, 13 met CC 1/2, 31 met CC 2, 22 met CC 2/3, 14 met CC 3, and 198 fully met all capability requirements. There were 24 F&Os issued to address the identified gaps to compliance with the PRA standard. Subsequent to the peer review, 13 of the findings have been addressed and 11 are still open pending the next model update. The open F&Os are listed in Table A-1 of Attachment 2 of this submittal, which includes what, if any, impact there may be to the ILRT extension.

- The 2012 fire PRA peer review for the PRA ASME model update identified 183 SRs to be reviewed for the GNPP PRA. Of these, 2 were not met, 2 met CC 1, 8 met CC 1/2, 17 met CC 2, 13 met CC 2/3, 7 met CC 3, 118 fully met all capability requirements, and 16 were not applicable. There were 19 findings and 22 suggestions issued to address potential gaps to compliance with the PRA standard. There were 3 Best Practices. All the findings that impact the fire PRA were closed prior to the initial NFPA 805 submittal. As the results of this peer review have already been communicated to the NRC as part of the

NFPA-805 submittal (Reference 19) and subsequent RAIs, these will not be catalogued in this document.

- A peer review was conducted to assess actions taken to address existing finding-level F&Os. The June 2017 Full Power Internal Event (FPIE) review performed an independent assessment of finding-level F&Os from previous peer reviews. Finding-level F&Os that were reviewed and were determined to have been adequately addressed through this technical review are considered "closed." These closed F&Os are no longer relevant to the current PRA model. The technical review team determined that 17 of the 23 finding-level F&Os were resolved. Four of the finding-level F&Os remain open. The remaining two finding-level F&Os were partially resolved but require further documentation (i.e., all technical aspects were resolved).

The remaining gaps are documented in the URE database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

3.4.2.5 Seismic PRA

The GNPP IPEEE seismic risk analysis did not quantify a CDF impact. The SCDF calculation is summarized in Section 5.2.7 and detailed in Appendix B of Attachment 2 of this submittal.

3.4.2.6 Consistency with Applicable PRA Standards

Based on the peer reviews, independent assessment of F&O resolutions, and the focused scope peer reviews, it is concluded that the current GNPP internal events and fire PRA models mostly conform to CC-2 of ASME RA-Sb-2009, ASME/ANS Standard for Probabilistic Risk Assessment of Nuclear Power Plant Applications as endorsed by RG 1.200 Revision 2 (with the remaining few items conforming to CC-I of ASME RA-Sb-2009). Since NEI 94-01 endorses using PRA models conformed to CC-I of the ASME/ANS standard, using these models for this ILRT analysis meets technical adequacy requirements.

3.4.2.7 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:

- RG 1.174 (Reference 4) provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in risk as resulting in increases of CDF 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 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 9.52E-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. Therefore, the estimated change in LERF is determined to be “very small” using the acceptance guidelines of RG 1.174 (Reference 4). The risk change resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years bounds the 1 in 10 years to 1 in 15 years risk change. Considering the increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 10 years to 1 in 15 years is estimated as 3.96E-8, the risk increase is “very small” using the acceptance guidelines of RG 1.174 (Reference 4).

- When external event risk is included, the increase in 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 3.91E-7/year using the EPRI guidance, and total LERF is 1.61E-6/year. As such, the estimated change in LERF is determined to be “small” using the acceptance guidelines of RG 1.174 (Reference 4). The risk change resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years bounds the 1 in 10 years to 1 in 15 years risk change. When external event risk is included, the increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 10 years to 1 in 15 years is estimated as 2.28E-7 and the total LERF is 1.45E-6. Therefore, the risk increase is “small” using the acceptance guidelines of RG 1.174 (Reference 4). As discussed in Sections 5.1.3 and 5.2.7, the EPRI methodology used to estimate the increase in LERF and the models used to estimate total LERF are conservative. Therefore, the conservative methodology adds margin.

- 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 0.29 person-rem/year. NEI 94-01, Revision 2-A (Reference 1) states that a 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 0.881%. NEI 94-01, Revision 2-A (Reference 1) states that an increase in CCFP of ≤ 1.5% is small. Therefore, this increase is judged to be small.

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

3.4.2.8 Previous Assessments

Historical ILRT extension evaluations provide further corroboration to support the conclusion that increasing the ILRT interval has only a small impact on plant risk. In NUREG-1493 (Reference 7), the NRC 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 GNPP confirm these general conclusions on a plant-specific basis considering the severe accidents evaluated for GNPP, the GNPP containment failure modes, and the local population surrounding GNPP.

3.4.3 RG 1.174, Defense-in-Depth Evaluation

RG 1.174, Revision 3, describes an approach that is acceptable for developing risk-informed applications for a licensing basis change that considers engineering and applies risk insights. One of the considerations included in RG 1.174 is defense-in-depth. Defense-in-depth is a safety philosophy that employs successive compensatory measures to provide accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The following seven considerations as presented in RG 1.174, Revision 3, Section 2.1.1.2 will serve to evaluate the proposed licensing basis change for overall impact on defense-in-depth.

1. *Preserve a reasonable balance among the layers of defense.*

A reasonable balance of the layers of defense (i.e., minimizing challenges to the plant, preventing any events from progressing to core damage, containing the radioactive source term, and emergency preparedness) helps to ensure an apportionment of the plant's capabilities between limiting disturbances to the plant and mitigating their consequences. The term "reasonable balance" is not meant to imply an equal apportionment of capabilities. The NRC recognizes that aspects of a plant's design or operation might cause one or more of the layers of defense to be adversely affected. For these situations, the balance between the other layers of defense becomes especially important when evaluating the impact of the proposed licensing basis change and its effect on defense-in-depth.

Response:

Several layers of defense are in place to ensure the GNPP containment structure, penetrations, isolation valves, and mechanical seal systems continue(s) to perform their intended safety function. The purpose of the proposed change is to extend the testing frequency of the Type A ILRT from 10 years to 15 years.

As shown in NUREG-1493, Performance-Based Containment Leak Rate-Test Program (Reference 7), increasing the test frequency of ILRTs up to a 20-year test interval 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. In addition, the study concluded that Types B and C tests could identify the vast majority (greater than 95 percent) of all potential leakage paths.

Several programmatic factors can also be cited as layers of defense ensuring the continued safety function of the GNPP containment pressure boundary. NEI 94-01, Revision 2-A requires sites adopting the 15-year extended ILRT interval to perform visual examinations of the accessible interior and exterior surfaces of the containment structure for structural degradation that may affect the containment leak-tight integrity at the frequency prescribed by the guidance; or, if approved through a TS amendment, at the frequencies prescribed by ASME Section XI. Additionally, several measures are put in place to ensure integrity of the Types B and C tested components. NEI 94-01 limits large containment penetrations such as airlocks, purge and vent valves, boiling water reactor (BWR) main steam and feedwater isolation valves, to a maximum 30-month testing interval. Therefore, the proposed change does not challenge or limit the layers of defense available to assess the ability of the GNPP containment structure to perform its safety function.

PRA Response:

The use of the risk metrics of LERF, population dose, and conditional containment failure probability collectively ensures the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. The change in LERF is “small” per RG 1.174, and the change in population dose and CCFP are “small” as defined in this analysis and consistent with NEI 94-01, Revision 3-A. This LAR was developed using the PRA standards referenced in NEI 94-01 Revision 3-A, as this is the most up to date guidance available.

2. *Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.*

Nuclear power plant licensees implement a number of programmatic activities, including programs for quality assurance, testing and inspection, maintenance, control of transient combustible material, foreign material

exclusion, containment cleanliness, and training. In some cases, activities that are part of these programs are used as compensatory measures; that is, they are measures taken to compensate for some reduced functionality, availability, reliability, redundancy, or other feature of the plant's design to ensure safety functions (e.g., reactor vessel inspections that provide assurance that reactor vessel failure is unlikely). NUREG-2122, "Glossary of Risk-Related Terms in Support of Risk-Informed Decision making," (Reference 20) defines "safety function" as those functions needed to shut down the reactor, remove the residual heat, and contain any radioactive material release.

A proposed licensing basis change might involve or require compensatory measures. Examples include hardware (e.g., skid-mounted temporary power supplies); human actions (e.g., manual system actuation); or some combination of these measures. Such compensatory measures are often associated with temporary plant configurations. The preferred approach for accomplishing safety functions is through engineered systems. Therefore, when the proposed licensing basis change necessitates reliance on programmatic activities as compensatory measures, the licensee should justify that this reliance is not excessive (i.e., not overly reliant). The intent of this consideration is not to preclude the use of such programs as compensatory measures but to ensure that the use of such measures does not significantly reduce the capability of the design features (e.g., hardware).

Response:

The purpose of the proposed change is to extend the testing frequencies of the Type A ILRT from 10 years to 15 years. Several programmatic factors were defined in the response to Question 1 above, which are required when adopting NEI 94-01, Revision 2-A. These factors are conservative in nature and are designed to generate corrective actions if the required testing or inspections are deemed unsatisfactory well in advance to ensure the continued safety function of the containment is maintained. The programmatic factors are designed to provide differing ways to test and/or examine the containment pressure boundary in a manner that verifies the GNPP containment pressure boundary will perform its intended safety function. Since the proposed change does not alter the configuration of the GNPP containment pressure boundary, continued performance of the tests and inspections associated with NEI 94-01 will only serve to ensure the continued safety function of the containment without affecting any margin of safety.

PRA Response:

The adequacy of the design feature (the containment boundary subject to Type A testing) is preserved as evidenced by the overall "small" change in risk associated with the Type A test frequency change.

3. *Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.*

As stated in RG 1.174 Rev. 3, Section C.2.1.1.1 , the defense-in-depth philosophy has traditionally been applied in plant design and operation to provide multiple means to accomplish safety functions.

System redundancy, independence, and diversity result in high availability and reliability of the function and also help ensure that system functions are not reliant on any single feature of the design. Redundancy provides for duplicate equipment that enables the failure or unavailability of at least one set of equipment to be tolerated without loss of function. Independence of equipment implies that the redundant equipment is separate such that it does not rely on the same supports to function. This independence can sometimes be achieved by the use of physical separation or physical protection. Diversity is accomplished by having equipment that performs the same function rely on different attributes such as different principles of operation, different physical variables, different conditions of operation, or production by different manufacturers, which helps reduce Common-Cause Failure (CCF). A proposed change might reduce the redundancy, independence, or diversity of systems. The intent of this consideration is to ensure that the ability to provide the system function is commensurate with the risk of scenarios that could be mitigated by that function. The consideration of uncertainty, including the uncertainty inherent in the PRA, implies that the use of redundancy, independence, or diversity provides high reliability and availability and also results in the ability to tolerate failures or unanticipated events.

Response:

The proposed change to extend the testing frequencies of the Type A ILRT from 10 years to 15 years does not reduce the redundancy, independence or diversity of systems. As shown in NUREG-1493, increasing the test frequency of ILRTs up to a 20-year test interval 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. Additionally, the study concluded that Types B and C tests could identify the vast majority (greater than 95 percent) of all potential leakage paths.

Therefore, the proposed change preserves system redundancy, independence, and diversity and ensures a high reliability and availability of the containment structure to perform its safety function in the event of unanticipated events.

PRA Response:

The redundancy, independence, and diversity of the containment subject to the Type A test is preserved, commensurate with the expected frequency and consequences of challenges to the system, as evidenced by the overall “small” change in risk associated with the Type A test frequency change.

4. *Preserve adequate defense against potential Common-Cause Failures (CCFs).*

An important aspect of ensuring defense-in-depth is to guard against CCF. Multiple components may fail to function because of a single specific cause or event that could simultaneously affect several components important to risk. The cause or event may include an installation or construction deficiency, accidental human action, extreme external environment, or an unintended cascading effect from any other operation or failure within the plant. CCFs can also result from poor design, manufacturing, or maintenance practices. Defenses can prevent the occurrence of failures from the causes and events that could allow simultaneous multiple component failures. Another aspect of guarding against CCF is to ensure that an existing defense put in place to minimize the impact of CCF is not significantly reduced; however, adding another defense can compensate for a reduction in one defense.

Response:

As part of the proposed change, GNPP will be required to adopt the performance-based testing standards outlined in NEI 94-01, Revision 2-A and ANSI/ANS 56.8-2002. NEI 94-01, Revision 2-A requires a cause determination be performed for those components that exceed their administrative leakage limits. The cause determinations will focus on those activities that can eliminate the identified cause of failure with appropriate steps to eliminate recurrence. One of the aspects of cause determinations focuses on common cause failures for components of a similar design, application and/or environment. As a result, adoption of the performance-based testing standards proposed by this change ensures adequate barriers exist to preclude failure of the containment pressure boundary due to common-mode failures and, therefore, continues to guard against CCF.

PRA Response:

Adequate defense against CCFs is preserved. The Type A test detects problems in the containment, which may or may not be the result of a CCF; such a CCF may affect failure of another portion of containment (i.e., local penetrations) due to the same phenomena. Adequate defense against CCFs is preserved via the continued performance of the Types B and C tests and the performance of inspections. The change to the Type A test, which bounds the risk associated with containment failure modes including those involving CCFs, does not degrade adequate defense as evidenced by the overall “small” change in risk associated with the Type A test frequency change.

5. *Maintain multiple fission product barriers.*

Fission product barriers include the physical barriers themselves (e.g., the fuel cladding, reactor coolant system pressure boundary, and containment) and any equipment relied on to protect the barriers (e.g., containment spray). In general, these barriers are designed to perform independently so

that a complete failure of one barrier does not disable the next subsequent barrier. For example, one barrier, the containment, is designed to withstand a double-ended guillotine break of the largest pipe in the RCS, another barrier.

A plant's licensing basis might contain events that, by their very nature, challenge multiple barriers simultaneously. Examples include ISLOCAs, steam generator tube rupture, or crediting containment accident pressure. Therefore, complete independence of barriers, while a goal, might not be achievable for all possible scenarios.

Response:

The purpose of the proposed change is to extend the testing frequencies of the Type A ILRT from 10 years to 15 years. As part of the proposed change, GNPP will be required to adopt the performance-based testing standards outlined in NEI 94-01, Revisions 2-A and ANSI/ANS 56.8-2002. The overall containment leakage rate calculations associated with the testing standards contain inherent conservatisms through the use of margin. Plant TS require the overall primary containment leakage rate to be less than or equal to $1.0L_a$. NEI 94-01 requires the as-found Type A test leakage rate must be less than the acceptance criterion of $1.0L_a$ given in the plant TS. Prior to entering a mode where containment integrity is required, the As-Left (AL) Type A leakage rate shall not exceed $0.75L_a$. The AF and AL values are as determined by the appropriate testing methodology specifically described in ANSI/ANS-56.8-2002. Additionally, the combined leakage rate for all Type B and Type C tested penetrations shall be less than or equal to $0.6L_a$, determined on a maximum pathway basis from the AL LLRT results prior to entering a mode where containment integrity is required. This regulatory approach results in a 25% and 40% margin, respectively, to the $1.0L_a$ requirements. For those local leak rate tested components that have demonstrated satisfactory performance and have had their testing frequencies extended, administrative testing limits are assigned on a component by component basis and are used to identify potential valve or penetration degradation. Administrative limits are established at a value low enough to identify and allow early correction in advance of total valve failure. Should a component exceed its administrative limit during testing, NEI 94-01, Revision 2-A states a cause determination should be performed in order to reinforce achievement of acceptable performance. The cause determination is designed to identify and address common-mode failure mechanisms through appropriate corrective actions. Therefore, the proposed change adopts requirements with inherent conservatisms to ensure the margin of safety is maintained, thereby, preserving the containment fission product barrier.

PRA Response:

Multiple Fission Product barriers are maintained. The portion of the containment affected by the Type A test extension is still maintained as an independent fission product barrier, albeit with a "small" change in the reliability of the barrier.

6. *Preserve sufficient defense against human errors.*

Human errors include the failure of operators to correctly and promptly perform the actions necessary to operate the plant or respond to off-normal conditions and accidents, errors committed during test and maintenance, and incorrect actions by other plant staff. Human errors can result in the degradation or failure of a system to perform its function, thereby significantly reducing the effectiveness of one of the layers of defense or one of the fission product barriers. The plant design and operation include defenses to prevent the occurrence of such errors and events. These defenses generally involve the use of procedures, training, and human engineering; however, other considerations (e.g., communication protocols) might also be important.

Response:

Sufficient defense against human errors is preserved. Human errors committed during testing and maintenance may be reduced by the less frequent performance of Type A tests (less opportunity for errors to occur).

PRA Response:

Sufficient defense against human errors is preserved. The probability of a human error where operators fail to correctly and promptly perform the actions necessary to operate the plant, or to respond to off-normal conditions and accidents is not significantly affected by the change to the Type A testing frequency. Errors committed during test and maintenance may be reduced by the less frequent performance of the Type A test (less opportunity for errors to occur).

7. *Continue to meet the intent of the plant's design criteria.*

For plants licensed under 10 CFR Part 50 or 10 CFR Part 52, the plant's design criteria are set forth in the current licensing basis of the plant. The plant's design criteria define minimum requirements that achieve aspects of the defense-in-depth philosophy; as a consequence, even a compromise of the intent of those design criteria can directly result in a significant reduction in the effectiveness of one or more of the layers of defense. When evaluating the effect of the proposed licensing basis change, the licensee should demonstrate that it continues to meet the intent of the plant's design criteria.

Response:

The purpose of the proposed change is to extend the testing frequencies of the Type A ILRT from 10 years to 15 years. The proposed extension does not involve either a physical change to the plant or a change in the manner in which the plant is operated or controlled. As part of the proposed change, GNPP will be required to adopt the performance-based testing standards outlined in NEI 94-01, Revision 2-A and ANSI/ANS 56.8-2002. The leakage limits imposed by plant TS remain unchanged when adopting the performance-based testing standards outlined in NEI 94-01, Revision 2-A,

and ANSI/ANS 56.8-2002. Plant design limits imposed by the UFSAR also remain unchanged as a result of the proposed change. Therefore, the proposed change continues to meet the intent of the plant's design criteria to ensure the integrity of the GNPP containment pressure boundary.

PRA Response:

The intent of the plant's design criteria continues to be met. The extension of the Type A test does not change the configuration of the plant or the way the plant is operated.

Conclusion:

The responses to the seven Defense-in-Depth questions above conclude that the existing defense-in-depth has not been diminished; rather, in some instances defense-in-depth has been increased. Therefore, the proposed change does not comprise a reduction in safety.

3.5 Non-Risk Based Assessment

Consistent with the defense-in-depth philosophy discussed in RG 1.174, GNPP has assessed other non-risk based considerations relevant to the proposed amendment. GNPP has multiple inspections and testing programs that ensure the containment structure remains capable of meeting its design functions and that are designed to identify any degrading conditions that might affect that capability. These programs are discussed below.

3.5.1 Safety-Related (Service Level 1) Coatings Program

The Safety-Related Coatings Program detailed in procedure IP-IIT-12, "Safety Related Coatings Program," is designed to provide added assurance of continued acceptable performance of coatings inside the containment. The program is intended to meet GNPP's commitment to the NRC to perform a monitoring program during each RFO.

A Containment coating condition assessment is conducted every outage to monitor and track the protective Service Level I coatings within the containment and report the results. The guidance is provided within the Containment Coating Condition Assessment Procedure identified as EP-3-P-0601. The program serves to ensure that coatings will not cause the ECCS to become inoperable during a DBA by overloading the Sump B strainers. The general scope of the assessment and coating work performed inside the Primary Containment includes:

- Qualitative inspection of accessible surfaces to assess the current condition of the protective coatings applied to areas of primary containment pressure boundary, structural steel, stairways and landings, piping, tanks, systems and components (valves, vessels and pumps), concrete walls, concrete floors and miscellaneous equipment.
- Visual inspection and evaluation of degraded coating and classification of deficiencies.

- Visual inspection of exposed substrates (if any) both steel and concrete to assess corrosion conditions.
- Photographic documentation of representative conditions.
- Assure all areas of identified loose flaking coatings are removed back to sound tightly adherent coating. Loose coating that cannot be reached will be reported to the coating engineer for future remediation and monitoring. This is required to limit the possibility of loose coatings that could potentially clog and restrict drains/strainer flow.

The inspection provides a survey of the condition of coatings inside containment. Deficient coatings found are recorded describing locations, types, quantities, and modes of failure. These are repaired within a reasonable time, and are evaluated to assure that there are no safety concerns. An overall report is issued to document the condition and assessment of the coatings.

3.5.1.1 Unqualified/Degraded Coatings in Containment

GNPP Site Engineering is responsible for verifying that the amount of unqualified coatings allowed in the primary containment does not exceed limits defined in design calculations and the UFSAR. A calculation is performed on the ECCS to ensure the amount of unqualified coatings in containment does not degrade the ECCS.

The total amount of protective coating debris (or Unqualified Coatings) inside the primary containment is limited to 6,300 square (sq.) ft. This quantity was considered acceptable based on ECCS suction strainer design analysis.

The estimated area of Unqualified Coatings inside the primary containment is following the two most recent RFOs summarized in Tables 3.5.1.1-1 and 3.5.1.1-2 below. The total area of unqualified coatings is still well below the design limit and is considered to be a conservative estimate.

Table 3.5.1.1-1 – Containment Unqualified Coatings following G1R39 RFO Fall 2015	
Design Limit	6,300 sq. ft.
Total area of degraded coatings following G1R39	1,015.12 sq. ft.
Total area of unqualified coatings following G1R39	1,441.50 sq. ft.
Total unqualified or degraded area following G1R39	2,456.62 sq. ft.
Percent of allowable degraded/unqualified coatings	39.0%

Table 3.5.1.1-2 – Containment Unqualified Coatings following G1R40 RFO Spring 2017	
Design Limit	6,300 sq. ft.
Total area of degraded coatings following G1R40	1,298.62 sq. ft.
Total area of unqualified coatings following G1R40	1,442.50 sq. ft.
Total unqualified or degraded area following G1R40	2,741.12 sq. ft.
Percent of allowable degraded/unqualified coatings	43.5%

3.5.2 Maintenance Rule Structural Assessment and Monitoring Program

The Structural Assessment and Monitoring Program was established to ensure the GNPP Maintenance Rule structural components are monitored and evaluated in accordance with the requirements of 10 CFR 50.65 using the guidance of the Nuclear Management and Resource Council (NUMARC) 93-01. The Maintenance Rule requires that licensees monitor the performance or condition of SSCs, against established criteria. Performance monitoring of structures is impractical; therefore condition monitoring has been set forth as the method of determining compliance with these established requirements. This program is applicable to structures and structural components and commodities used in meeting the regulatory requirements of the maintenance rule (10 CFR 50.65) and the license renewal rule (10 CFR 54).

This program shall be used to establish the initial baseline assessment of plant structures and all subsequent assessments performed thereafter. The information compiled in the baseline assessment shall serve as the basis for the scope of subsequent assessments. The frequency at which periodic structural examinations are conducted should be directly related to the condition and safety significance of the specific structure. For GNPP structures, following the initial baseline inspection and where no degradation or defects were identified by the inspection, a five-year inspection interval shall be used. For structures where evidence of degradation has been identified during the baseline inspection, the recommended action from the baseline inspection shall be completed within one year of the initial finding. Follow-up inspections will be scheduled and the frequency will be adjusted as part of the disposition of the Condition Report (CR) that is initiated when degradation is identified.

This program is applicable to SSCs and commodities, which are inaccessible during normal plant operations. These SSCs shall be inspected during plant RFOs with inspection frequencies consistent with this procedure. For example, Containment interior structures (inaccessible during plant operations) and normally high radiation areas during plant operations in the Auxiliary Building. When inaccessible areas of the containment structure, as defined in the ASME BPV Section XI Code, Subsections IWE and IWL, become accessible, an inspection of those areas shall be performed and included in this program’s database.

The following are areas for SSCs inside of containment in the scope of License Renewal, which credit the Structural Assessment and Monitoring Program for Aging Management.

a. Internal reinforced concrete components including beams, floor slabs, shield walls, secondary compartment walls, refueling cavity walls, equipment pads, hatch blocks, missile shields, curbs, and miscellaneous features.

b. Structural steel floor framing, overhead crane support columns and girders, grating, stairs, platforms and access ladders.

c. Support anchorages, pipe whip restraints, embedded steel and component supports.

d. Reactor disassembly lift fixtures, support stands and refuel support mechanisms.

e. Equipment sumps and sump screen components.

The interior of containment contains electrical panels, enclosures, conduits and cable trays, instrument lines, fuel handling equipment and miscellaneous overhead handling systems.

Areas exempt from this inspection include:

- The Containment Liner credits the ASME Section XI Inservice Inspection (ISI) IWE program.
- Penetration 29 credits the ASME Section XI ISI IWE program.
- “B” sump inspection (non-accessible portion) credits the GNPP Work Planning Recurring Task Scheduler.
- All Class 1, 2 and 3 SSCs credit ASME Section XI IWB, IWC and IWD; however, it should be reported to the ISI Engineer if issues are discovered.

3.5.3 Containment Inservice Inspection Plan

The GNPP Containment Inservice Inspection (CISI) Plan is established in accordance with 10 CFR 50.55a. This plan has been developed to comply with the ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, and implements the requirements of:

- UFSAR 3.8.1.7.3.6, Inservice Inspection
- UFSAR 3.8.1.7.3.3, Current Tendon Surveillance Program
- UFSAR 18.2.1.3, ASME, Section XI, Subsections IWE & IWL Inservice Inspection
- Technical Specification 3.6.1, Containment Systems
- Technical Specification 5.5.6, Pre-Stressed Concrete Containment
- Tendon Surveillance Program

The Second Ten-Year CISI Plan also conforms to the latest revision of GNPP Station License Renewal Aging Management Program Basis, ASME Section XI, Subsections IWE & IWL Inservice Inspection Program, LR-IWEL PROGPLAN.

ASME Section XI Code of Record for the Second Ten-Year CISI Interval

10 CFR 50.55a requires that inservice inspection of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference 12 months prior to the start of the 120-month inspection interval.

On October 24, 2008, the edition of the ASME Code Section XI, Subsections IWE and IWL, accepted by the NRC was the 2004 Edition, no Addenda. Subsections IWE and IWL contain the requirements for CISI of Class MC components (metallic containments and metallic shell and penetration liners of Class CC containments)

and Class CC components (concrete containments) of light-water cooled nuclear power plants.

The CISI Plan for ASME Class MC and CC components for the Second Ten-Year CISI Interval has been developed using the ASME Code, Section XI, 2004 Edition; except where specific written alternatives from Code requirements has been requested by EGC and granted by the NRC or as amended by the NRC in 10 CFR 50.55a.

3.5.3.1 Inspection Intervals

IWE (Class MC) Inspection Interval and Periods

The first IWE and IWL CISI Interval was developed to meet the expedited requirements identified in 10 CFR 50.55a to be implemented by September 9, 2001. The first IWE and IWL intervals at GNPP were scheduled to end September 9, 2008. However, the interval for both was extended per Relief Request (RR) No. 22 (Reference 30) until December 31, 2009; so that the CISI interval would coincide with the ISI interval, which was already established.

The first IWL Tendon Inspection Program and concrete examination was completed prior to September 1, 2001. For future scheduling purposes, the first examinations will be referred to as the year 2000 (+/- 1 yr.).

The Second Ten-Year CISI interval for the performance of Containment ISI complies with IWE-2412, Inspection Program B, and began on January 1, 2010, and will end on December 31, 2019.

It should be noted that the second CISI interval concludes at the end of 2019. The next ILRT is currently scheduled for April 2021, which is after the end of the second CISI interval. The third ten-year CISI Plan is currently under development and begins in January 2020. The code of record for the next ten-year CISI interval will be the 2013 edition of ASME Section XI.

The dates for the Second Ten-Year CISI Interval are prescribed by the NRC in the 5th RR No. 22 (Reference 30). These dates coincide with the GNPP 5th ISI Interval. The three periods, within the interval, are defined in Section XI and are provided in Table 3.5.3.1-1, as follows:

Table 3.5.3.1-1 – GNPP Fifth Containment MC/CC Inspection Interval

First Period	Second Period	Third Period
1/1/2010 – 12/31/2012	1/1/2013 – 12/31/2016	1/1/2017 – 12/31/2019
Outage 1 (G1R36)	Outage 3 (G1R38)	Outage 5 (G1R40)
Outage 2 (G1R37)	Outage 4 (G1R39)	Outage 6 (G1R41)

IWL (Class CC) Inspection Periods

This inspection is effective for IWL inspections conducted between January 1, 2010, and December 31, 2019.

Concrete examinations shall be conducted at five-year intervals (+/- one year) as described in IWL-2410(a) and (c). For the purposes of the CISI Plan, an IWL inspection period shall commence not more than 1 year prior to the specified dates and shall be completed not more than one year after such dates. If plant operating conditions are such that examination of portions of the concrete cannot be completed within this stated time interval, examination of those portions may be deferred until the next regularly scheduled plant outage.

Concrete surface areas affected by a repair / replacement activity shall be examined at one year (± three months) following completion of repair/replacement activity. If plant operating conditions are such that examination of portions of the concrete cannot be completed within this time interval, examination of those portions may be deferred until the next regularly-scheduled plant outage.

The requirements of IWL-2410(b) (one, three, and five-year examinations) do not apply to GNPP because more than five years have passed since the completion of the initial Structural Integrity Test (SIT). The SIT for initial operation of GNPP was completed in April 1969.

The initial IWL inspections were completed to comply with the expedited requirements identified in 10 CFR 50.55a for the examinations to be implemented by September 9, 2001. These examinations serve the same purpose as preservice examinations for newly constructed plants and are considered to be “baseline” examinations.

For GNPP, the time frame for completing the expedited examinations did not correspond with a five-year interval from the original SIT date. As a result, GNPP IWL inspection periods are based on the completion date of the examinations conducted for the first interval.

Examinations of un-bonded post-tensioning systems shall be conducted at five-year intervals as described in IWL-2420(a) and (c).

The resulting IWL periods are shown in the following Table 3.5.3.1-2, in which the current interval exams are shown in **bold**:

Table 3.5.3.1-2 – GNPP IWL Examination Periods

Period	Date	Tolerance
30 Year	9/10/2000	+/- 1 Year
35 Year	9/10/2005	+/- 1 Year
40 Year	9/10/2010	+/- 1 Year
45 Year	**9/10/2015**	**+/- 1 Year**
50 Year	**9/10/2020**	**+/- 1 Year**
55 Year	9/10/2025	+/- 1 Year
60 Year	9/10/2030	+/- 1 Year

Code Cases Approved Through Request for Alternatives

There are no additional CISI code cases approved for use at GNPP through a request for alternatives, outside of those ASME Code Cases contained in RG 1.147, Revision 17.

Code Cases Adopted Via 10 CFR 50.55a

There are no containment related ASME Code Cases that are not contained in Regulatory Guide 1.147, Revision 17, but, are mandated in 10 CFR 50.55a as augmented requirements.

3.5.3.2 Application Criteria and Code Compliance

Examination Categories

The following provides a summary of the application of ASME Code, Section XI, 2004 Edition to the GNPP Second Ten-Year CISI Interval Program. The application and distribution of examinations for this interval is based upon utilizing Inspection Program B as defined by Articles IWE-2412 and the inspection interval as described in IWL-2400 of Section XI. The results of this application are summarized by ASME Category and Item Number and are contained within Tables 3.5.3.2-3, 3.5.3.2-4, and 3.5.3.2-5. These tables only contain those ASME Item numbers that are relevant to GNPP.

Table 3.5.3.2-3 – GNPP Code Category Summary

Category	Item Number	Description	Exam Method	Number of Components in Item No.	Required to be Examined During Interval	Examination Percentage Required	Number to be Examined in Interval	Number to be Examined in First Period	Number to be Examined in Second Period	Number to be Examined in Third Period
E-A	Containment Surfaces									
E-A	E1.11	Containment Vessel Accessible Surface Areas	General Visual	129	387(2)	100%	Each Inspection Period	129	129	129
E-A	E1.11b(1)	Bolted Connections	VT-3	55	55	100%	Each Inspection Interval	18	18	19
E-A	E1.12	Wetted Surfaces of Submerged Areas	General Visual	1	1	100%	Each Inspection Interval	1	0	0
E-A	E1.30	Moisture Barriers(3)	General Visual	4	11(2)	100%	Each Inspection Period	3	4	4
Category Total				189	454(2)			151	151	152
Notes for Cat. E-A	Note 1: Item Number E1.11b is utilized to ensure compliance with the specific requirements of 10 CFR 50.55a(b)(2)(ix)(G) and (H) for bolting. Note 2: Examination of this item number is required each period. Therefore, the number required during the interval is three times the total number of components. This is also reflected in the category total. Note 3: Added leak Chase Channel Caps per NRC IN 2014-07, applicable to 2nd and 3rd periods.									

Table 3.5.3.2-4 – GNPP Code Category Summary

Category	Item Number	Description	Exam Method	Number of Components in Item No.	Required to be Examined During Interval	Examination Percentage Required	Number to be Examined in Interval	Number to be Examined in First Period	Number to be Examined in Second Period	Number to be Examined in Third Period
E-C	Containment Surfaces Requiring Augmented Examinations									
E-C	E4.11	Visible Surfaces	VT-1	3	9[1]	100%	Each Inspection Period[2]	3	3	3
E-C	E4.12	Surface Area Grid Minimum Wall Thickness Location	Ultrasonic Thickness	3	9[1]	100%	Each Inspection Period[2]	3	3	3
Category Total				6	18[1]			6	6	6
Notes for Cat. E-C	Note 1: Examination of this item number is required each period. Therefore the number required during the interval is three times the total number of components. This is also reflected in the Category Total. Note 2: Examination of this item number is required each period until the areas remain essentially unchanged for one period.									
L-A	Concrete									
L-A	L1.11	All accessible surface areas	General Visual	18	36[1]	Essentially 100% (see IWL-2510)	Twice per Interval (every 5 years)	18	18	0
Category Total				18	36[1]			18	18	0
Notes for Cat. L-A	Note 1: IWL-2410 requires 100% examination every five years. Therefore the number required during the interval is twice the total number of components. This is also reflected in the Category Total.									

Table 3.5.3.2-5 – GNPP Code Category Summary

Category	Item Number	Description	Exam Method	Number of Components in Item No.	Required to be Examined During Interval	Examination Percentage Required	Number to be Examined in Interval	Number to be Examined in First Period	Number to be Examined in Second Period	Number to be Examined in Third Period
L-B	Unbonded Post-Tensioning System									
L-B	L2.10	L-B	IWL-2522	160	8[(1)]	100%	IWL-2420	4[(2)]	4[(2)]	0
L-B	L2.20	L-B	IWL-2523.2	160	2[(1)]	100%	IWL-2420	1[(3)]	1[(3)]	0
L-B	L2.30	L-B	Detailed Visual	160	8[(1)]	100%	IWL-2420	4[(2)]	4[(2)]	0
L-B	L2.40	L-B	IWL-2525.2(a)	160	8[(1)]	100%	IWL-2420	4[(2)]	4[(2)]	0
L-B	L2.50	L-B	IWL-2525.2(b)	160	8[(1)]	100%	IWL-2420	4[(2)]	4[(2)]	0
			Category Total	800	34			17	17	0

Notes for Cat. L-B

Note 1: IWL-2410 requires 100% examination every five years. Therefore the number required during the interval is twice the requirement. This is also reflected in the Category Total.

Note 2: The 4 tendons selected include the following: one Common and three Random.

Note 3: The L2.20 exams do not apply to Common Tendons per IWL-2521(b). There is only one type of tendon: vertical, therefore only one L2.20 exam.

3.5.4 Supplemental Inspection Requirements

With the implementation of the proposed change, TS 5.5.15 will be revised by replacing the reference to RG 1.163 (Reference 3) with reference to NEI 94-01, Revision 2-A (Reference 1). This will require that a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity be conducted. 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 2-A:

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

In addition to the IWE and IWL examinations scheduled in accordance with the CISI Program, the performance of inspections in accordance with the Appendix J Primary Containment Inspection will be utilized to ensure compliance with the visual inspection requirements of NEI 94-01, Revision 2-A. These inspections are conducted in accordance with STP-O-R-6.5.

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

GNPP Types B and C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and CIVs in accordance with 10 CFR 50, Appendix J, Option B and RG 1.163 (Reference 3). 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 acceptable limits. In accordance with GNPP TS 5.5.15, the allowable maximum pathway total Types B and C leakage is $0.6L_a$ (109,183 standard cubic centimeters per minute (SCCM)) where L_a equals approximately 181,971 SCCM).

As discussed in NUREG-1493 (Reference 7), 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 2008 through 2018 for GNPP has shown substantial margin between both the actual As-Found and As-Left outage leakage rate summations and the regulatory requirements as described below:

- The As-Found minimum pathway leakage rate for GNPP shows an average of 5.25% of 0.6 L_a with a high of 13.49% 0.6 L_a.

- The As-Left maximum pathway leakage rate for GNPP shows an average of 6.18% of 0.6 L_a with a high of 7.73% 0.6 L_a.

Table 3.5.5-1 provides a LLRT data trend summary for GNPP inclusive of the 2011 ILRT.

Table 3.5.5-1 – GNPP Type B and C LLRT Combined As-Found / As-Left Trend Summary

Year	2008	2009	2011	2012	2014	2015	2017	2018
RFO	G1R34	G1R35	G1R36	G1R37	G1R38	G1R39	G1R40	G1R41
AF Min Path (cc/min.)	3766	3098	14726	3348	5050	5103	5600	5153
Fraction of 0.6 La (percent)	3.45	2.84	13.49	3.07	4.63	4.68	5.13	4.72
AL Max Path (cc/min.)	4797	5316	7789	6727	5897	8437	7854	7122
Fraction of 0.6 La (percent)	4.40	4.87	7.14	6.17	5.41	7.73	7.20	6.52
AL Min Path (cc/min.)	3799	3062	3620	2649	3689	4989	5852	5173
Fraction of 0.6 La (percent)	3.48	2.81	3.32	2.43	3.38	4.57	5.36	4.74

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

No Type B or Type C components on an extended test frequency have exceeded their administrative leakage limits over the last five RFOs at GNPP.

GNPP Types B and C Component Performance:

The percentage of the total number of GNPP Type B tested components that are on 60-month extended performance-based test intervals is 64.0%.

GNPP elected to not place any Type B penetrations on the 120-month interval, and thereby, did not adopt the maximum test frequency allowed under NEI 94-01, Revision 0, Section 11.3.2. The GNPP test frequency for Type B penetrations is 60 months. Those Type B penetrations not on a 60-month test frequency are either:

a) used during RFOs and therefore must be AL tested each RFO subsequent to use, or

b) limited to a test frequency of 30 months per TS 5.5.15.

The percentage of the total number of GNPP Type C tested components that are on 60-month extended performance-based test intervals is 76.04%.

The Type C penetrations not on a 60-month test frequency are either:

a) on a 30-month frequency following valve replacement or major maintenance to re-establish their performance history of two satisfactory sequential AF tests, or
b) used or removed during RFOs to support Flex or outage requirements;
c) Tested on a RFO frequency to satisfy IST 24-month test frequency requirements.

3.6 Operating Experience (OE)

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.

The following site specific and industry events have been evaluated for their impact on GNPP’s primary containment:

- 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 3.6.1 through 3.6.5, respectively.

3.6.1 IN-1992-20, "Inadequate Local Leak Rate Testing"

The NRC issued IN 92-20 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 Nuclear Power Station, 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 was designed. This omission led to a leakage path that was not discovered until the plant performed an ILRT.

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 plant should have performed Type C LLRT on the valves.

Discussion:

IN 1992-20 is not applicable to GNPP. There are no two ply bellows on containment penetrations at GNPP that perform a containment isolation function. The bellows are single-ply, American Society of Testing and Materials (ASTM) A240, Type 304 stainless steel and function to accommodate lateral and axial pipe displacements. Prior to the performance of Type A testing, the penetration bellows are aligned to their associated mechanical manifolds to permit the monitoring of the containment primary barrier welds. The pressure gauge for each manifold will be monitoring a group of penetrations. The manifolds exhibiting pressure build-up during the ILRT, and the penetrations served by those manifolds, will be individually checked upon completion of the ILRT and the leakage located and the leak rate determined. In addition, local leak rate testing methods have been verified to account for all possible leakage paths, including those through gasketed flanges.

3.6.2 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:

There have been numerous industry events and NRC INs relative to containment liner corrosion. The root cause of the containment liner issue is exposure of the metal liner to water/fluids, etc. For GNPP the principal cause of exposure of the liner to water is due to the leakage of the refueling cavity during plant shutdowns. The borated water from this leakage migrates down to the basement level and also into Sump A.

Disposition:

The containment structural concrete and liner are ASME Code components and are covered and monitored by GNPP’s Inservice Inspection (ISI) Program. Monitoring of the containment structure/liner will be conducted into the future through that program. Additional inspections as determined to be appropriate by System Engineering will be performed during future RFOs.

Closure Summary:

Based on work activities completed during the 2000 RFO, certain areas of the liner that were repaired at that time were inspected during the 2005 RFO. The containment concrete and liner will be inspected on scheduled intervals under GNPP’s ISI Program. System Engineering will also inspect the material condition of the containment structure during future RFOs and subsequent to plant start up.

3.6.3 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 is applicable to the containment liner and concrete structure at GNPP. GNPP maintains sufficiently robust barriers and programs to prevent a similar condition. The ISI Program provides for frequent and periodic examinations of both the steel containment liner and the concrete structure. In addition, the performance of Type A leakage testing, in accordance with 10 CFR 50, Appendix J, Option B, serves to verify the leak tightness of the containment liner and ensure that no leakage pathways exist that could challenge the integrity of the containment structure. No gaps in the barriers at GNPP have been identified. The existing programs and barriers provide adequate protection for the issue identified in IN 2010-12. No existing programs are impacted or warrant revision.

3.6.4 <u>IN 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner"</u>

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 basemat metallic shell and liner plate seam welds of PWRs are embedded in 3 ft. by 4 ft. 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 basemat 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-in. 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, 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:

IN 2014-07 identifies degraded accessible components of their containment liner weld leak test channels. Degraded test port caps or box covers have the potential to allow moisture intrusion to embedded and inaccessible areas of the pressure retaining steel liner that can cause corrosion degradation. It also identified that the subject power stations did not have a program in place to periodically inspect these accessible test channel components as part of their CISI program.

The containment liner at GNPP is equipped with leak channels over the seam welds to facilitate weld testing during construction. The leak channels embedded in the basement floor slabs had their test ports extended to the floor surface using extension pipes. The test ports were capped using threaded pipe plugs after testing.

The test ports for the test channels at GNPP are susceptible to damage or degradation similar to the test ports at other stations identified within IN 2014-07. Should one of the pipe plugs become damaged or degraded, there is a potential that an intrusion of moisture could occur, which could potentially cause corrosion degradation of the embedded, inaccessible, pressure retaining boundary of the containment structure. In addition, similar to the other station identified within IN 2014-07, GNPP is not inspecting the condition of these test plugs on a periodic basis in accordance with the station's CISI program. The NRC considered these test ports as performing the same function as a "moisture barrier" component of the containment structure as they perform the same function. The function of a "moisture barrier" component is to prevent intrusion of moisture, which could cause corrosion degradation of inaccessible embedded pressure retaining components of the structure. Therefore, in accordance with ASME Section XI, Subsection IWE, Table IWE-2500-1, these components require periodic general visual examination, which has not been the case at GNPP.

The CISI program and aging management review did not identify the weld test channel test ports as "moisture barrier" components, nor did GNPP recognize the potential for a degraded test port cap to have the potential to cause corrosion degradation of the embedded steel liner.

The CISI program periodic inspections required revision to include inspection of the accessible test channel test ports so that any degradation, which has the ability to impact embedded portions, is identified.

This was corrected by revising the Second Interval CISI Plan to include floor welded leak chase channel caps to Category E-A, Item Number E1.30 applicable

to the second and third periods. Code examinations were performed and found to be acceptable.

3.6.5 NRC 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 a 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:

GNPP is currently in the second 10-year interval of the Containment Inservice Inspection Program and has been performing moisture barrier examinations in accordance with ASME Section XI Code, Item Number E1.30 as required by 10 CFR 50.55a since the start of the containment program in September 2001, even though the moisture barrier was not specifically identified within design documents. GNPP performs moisture barrier examinations on the caulking that seals the concrete basement floor to the stainless steel facing that is over the sealed rigid insulation panels that protect the containment metal liner. The outside of the GNPP Containment is concrete with no exposed metal liner.

No other additional actions are needed at this time. This item should be considered complete and closed at this time. No specific action or written NRC response is required for NRC RIS 2016-07.

3.6.6 Results of Recent Containment Inspections

Primary Containment Coatings Condition Assessment RFO G1R39 Fall 2015

Underwater Engineering Services, Inc. (UESI) completed the coating assessment of the Service Level I Primary Containment coated surfaces at GNPP during the 2015 G1R39 RFO.

Interior surfaces of the primary containment, components and equipment were inspected and assessed. The coating assessment commenced on October 19, 2015, at the basement elevation and was completed October 28, 2015, in the "B" sump. The "A" sump was inspected simultaneously with the Basement Level.

Inspection Findings:

The containment liner wall is protected with Carboline CZ-11 Zinc Rich Primer. The dome and a 3 ft. X 15 ft. section of the liner wall in the basement are the only areas of the liner wall that were accessible for inspection. Insulation panels were temporarily removed exposing the 3 ft. X 15 ft. section of liner wall at the basement level to provide access for an IWE inspection. Non-Destructive Examination (NDE) personnel inspected the dome liner wall from the operation level via remote means, using flashlights and binoculars. Remaining areas of the interior liner wall are covered with precision cut insulation panels. The panels are attached to the liner wall from the basement elevation to approximately 30 ft. above the operations floor elevation. Areas of liner plate covered with insulation panels are exempt from this coating assessment; the wall is not visible without removal of the panels.

The overall coating system of the interior surfaces of containment building appears to be generally in fair to good condition. Small to medium-sized, isolated to random coating defects were identified on components, pipes, internal structures and concrete floors and walls. Coating defects include: mechanical damage, burnt coatings, cracked/flaking coating, delamination, pinpoint rusting, blistering and checking (i.e., hair line cracks in coating that do not penetrate to substrate). Checking coating was mainly identified on concrete walls that appear to have been recoated. The majority of coating deficiencies are the result of mechanical damage. This is not age-related degradation; however, it is still a significant coating issue that must be monitored and remediated as deemed necessary by the site coating engineer. The exposed carbon steel areas are exhibiting uniform surface corrosion. No evidence of pitting corrosion was observed. Accessible loose or flaking coating is removed each outage by maintenance workers to mitigate sump/drainage clogging

Basement Level:

The coating system at the basement level/elevation, including the two loops (A & B) appears to be generally in fair to good condition. Small to medium-sized, isolated to random coating defects were identified on pipes, internal structures, handrails and stair landings, concrete floors and walls. Coating defects include: mechanical damage, burnt coatings, delamination, flaking coating, and checking. Checking coating was mainly identified on concrete walls that appear to have been recoated. Exposed carbon steel areas exhibited uniform surface corrosion. No evidence of pitting corrosion was observed. The majority of coating deficiencies (mechanical damage) are located on the basement floor. Numerous gouges were identified in the concrete floor. Significant coating deficiencies were numbered with a permanent marker, photographed, and placed in a database for future monitoring and remediation. Gouges in concrete appear to be the result of mechanical damage caused by equipment movement and scaffolding impact.

Several insulation panels were removed from the liner wall at the concrete floor intersection (moisture barrier) at the basement elevation. A 3 ft. X 15 ft. section of liner wall/moisture barrier was exposed for NDE inspection. The inorganic zinc applied to the liner wall has acted as an anode and was galvanically sacrificed in random locations approximately 4 in. up the liner wall, exposing bare substrate

that is heavily rusted in some areas. Pinpoint rusting was also observed in some areas where the inorganic zinc was sacrificed, protecting the liner wall carbon steel substrate.

Coating deficiencies identified during previous inspections were revisited and do not appear to have changed significantly. Areas of loose top coating that can be accessed without erecting scaffolding is systematically removed back to tightly adherent coating during each outage.

Coating inside (A) sump is in good condition. Nine (9) coating deficiencies, which were identified during a previous inspection, were re-inspected and determined not to have changed. The aforementioned coating deficiencies were not repaired due to limited outage scheduling. Several other laminated/flaking coatings areas inside A-sump were identified. The areas were photographed and documented for future remediation.

The walls and floor inside (B) sump are constructed of concrete and are not coated. The concrete is in good condition. The ceiling located inside the sump, a tank and a pipe appear to be constructed of galvanized steel and are in good condition. The interior surface of a carbon steel penetration embedded in a concrete wall exhibits uniform rusting. The corrosion is not significant. The ladder rungs inside the sump are constructed of carbon steel and are covered with uniform rust.

Intermediate Level:

The coating system at the intermediate level/elevation appears to be generally in fair to good condition. Small to large sized, isolated to random coating defects were identified on components, pipes, internal structures and concrete floors and walls. Coating defects include; mechanical damage, burnt coatings, flaking coating, delamination, and checking coatings. Checking coating was mainly identified on concrete walls. It appears the checking coating was applied over an existing coating. Documentation to validate past recoating activities was not provided. Several carbon steel components were installed without protective coating applied. The components exhibit general surface corrosion. No evidence of pitting corrosion was observed. Coating defects on steel and concrete were documented and prioritized for future monitoring and remediation.

Operation Level:

The coating system at the operations level appears to be generally in good condition. The liner wall was not inspected at this level due to installed insulation panels. Small to medium sized, isolated to random coating defects were identified on components, pipes, internal structures and concrete floors. Coating defects include; mechanical damage caused by equipment and material handling, burnt coatings caused by grinding and welding operation, flaking coatings, and checking coatings. Checking coating was mainly identified on concrete walls. Exposed carbon steel areas exhibited general surface corrosion. No evidence of pitting corrosion was observed. Coating deficiencies were documented and prioritized for future monitoring and remediation.

Containment Dome:

The dome is coated with inorganic zinc. Unlike other areas of the liner wall, the dome is not covered with insulation panels. The site NDE department inspected the inorganic zinc on the dome remotely, using spotlights and binoculars. The inorganic zinc appears to be in fair to good condition. However, numerous localized areas of pinpoint rusting were identified and photographed. Pinpoint rusting is generally caused by insufficient millage applied to a substrate during initial application or the sacrificial deteriorating process of the primer, thus, exposing substrate peaks. Generally, the primer appears to be in fair to good condition.

Discussion and Summary:

UESI coating inspectors performed an inspection and assessment of the protective coatings applied to structures and components on the interior surfaces of GNPP primary containment. Specific emphasis was placed on coatings applied to accessible areas of the pressure boundary (liner wall). The objective of the assessment is to identify newly degraded coating and/or apparent unqualified coatings, quantify the extent of these conditions and make a comparative assessment with previous inspection data. Inspection data collected will assist site engineering in determining: (1) the effects of degraded coating on plant operation, to ensure that the ECCS and the safety-related Containment Spray System (CSS) remain capable of performing their intended safety functions; and to mitigate corrosion of the containment liner and its integral structural and mechanical components. Coating repair work was performed during G1R39 on four (4) components located inside the primary containment. The components are as follows: Penetrations 312 and 323; and Compartment Cooler-A & B heat exchangers, Service Water inlet/outlet.

Conclusions and Recommendations:

The overall coating system inside the primary containment building is in fair to good condition. The coating assessment identified areas of coating degradation that should be repaired to mitigate corrosion. Typical degraded areas are identified above. No current coating conditions were observed that could impact structural integrity, plant operations, or safe shutdown. There is a significant amount of mechanical damage to the concrete floors throughout the containment building. Coating repair work should be planned and scheduled during future outages to address the areas of coating degradation identified during G1R39 and previous outages. An effectively executed repair plan will reduce radiation levels resulting from fixed contamination on exposed substrates, (both steel and concrete) mitigate progressive coating degradation by removing loose coating back to tightly adherent coating and making required safety-related coating repairs each RFO. Insulation panels should be temporarily removed from the liner wall in the basement level as early as practical in the outage to inspect the liner wall/concrete floor transition (moisture barrier). Pending inspection results, painters will have sufficient time to repair coating deficiencies identified without impacting the outage schedule. The unqualified coatings margin was well below the design limit against the ECCS strainer. Following G1R39, the unqualified coatings as a percentage of the design limit was 39.9%.

Primary Containment Coatings Condition Assessment RFO G1R40 Spring 2017

UESI completed the coatings assessment of the Service Level I Primary Containment coated surfaces at GNPP during the 2017 G1R40 RFO.

Interior surfaces of the primary containment, components and equipment were inspected and assessed. This coating assessment commenced on April 25, 2017, at the basement level and was completed April 28, 2017, in the Bravo ("B") loop.

Inspection Findings

The containment liner is protected with Carboline CZ-11 Zinc Rich Primer. The interior dome and two sections of the liner located in the basement are the only areas of the liner that were accessible for visual inspection. The remaining containment liner is covered with insulation panels. Several insulation panels were temporarily removed from two sections of liner at the basement level/moisture barrier intersection to provide access for an ASME Section XI, IWE examination, which was performed by others. The containment dome area was inspected remotely, from the top of the steam generator platforms. High power flashlights were used to perform the inspection. The remaining areas of containment liner are covered with precision cut insulation panels. The panels are attached to the liner starting at the basement level moisture barrier and continue up the containment wall to approximately 30 ft. above the operating floor level. Areas of liner covered with insulation panels are exempt from this coating assessment; the wall is not visible without removing the panels.

The protective coating system on the interior surfaces, components and piping of primary containment appear generally to be in fair to good condition. Twenty-three (23) new coating deficiencies (from deficiencies identified in previous inspections) were identified, marked, and reported to the Responsible Site Coating Engineer for tracking. The new locations are generally small (1 sq. in.) to medium (5 sq. ft.) in size, and the frequency is localized to random. Degraded coating areas were identified on exposed areas of liner wall, components, pipes, internal structures and concrete floors and walls. Numerous coating deficiencies were identified during previous walk down inspections and have not significantly changed. Coating defects include: mechanical damage, burnt coatings, cracked/flaking coating, delamination, pinpoint rusting, blistering and checking. Checking coating was mainly identified on concrete walls that appear to have been recoated.

The majority of coating deficiencies are the result of mechanical damage. This is not age-related degradation; however, it is still a significant coating issue that must be monitored and remediated as deemed necessary by the site coating engineer. Exposed carbon steel substrate areas exhibit uniform surface corrosion. No evidence of pitting corrosion was observed. Accessible loose or flaking coating is removed each outage by sub-contractor painters to mitigate sump/drainage clogging. Coating repairs completed during previous outages are in good condition.

Several areas of suspected unqualified coatings were identified in “B” loop approximately 35 ft. from the basement floor. Subject coating is applied on the existing top coat of several walls and totals 200 to 300 sq. ft.

Basement Level

The containment coating system at the basement level elevation, including the two loops (A & B) appear to be generally in fair to good condition. Small to medium-sized, localized to randomly-distributed coating defects were identified on pipes, internal structures, handrails and stair landings, concrete floors and walls. Coating defects include: mechanical damage, burnt coatings, delamination, flaking coating, cracking and checking coatings. Checking coating was mainly identified on concrete walls and floors that appear to have been recoated. Exposed carbon steel substrate areas exhibit uniform surface corrosion. No evidence of pitting corrosion was observed. The majority of coating defects (mechanical damage) are located on the basement floor. Numerous gouges were identified in the concrete floor. Significant coating deficiencies were numbered with a permanent marker, photographed, and placed in a database for future monitoring and remediation. Gouges in concrete appear to be the result of mechanical damage caused by equipment relocation/staging and scaffolding impact.

Several insulation panels were removed from the liner in two locations at the basement level concrete floor (moisture barrier). The areas where panels were moved for NDE inspections measured 3 ft. X 15 ft. and 3 ft. X 12 ft., respectively. The inorganic zinc primer applied to the containment liner, in areas where insulation was removed is depleted in several locations, exposing bare substrate with surface corrosion. Pinpoint rusting was also observed in some areas where the inorganic zinc primer was depleted.

Coating deficiencies identified during previous inspections were revisited and do not appear to have significantly changed. Areas of loose top coating that can be accessed without erecting scaffolding is systematically removed back to tightly adherent coating during each outage.

Coating inside (“A”) sump has not changed and is in good condition. Nine (9) coating deficiencies that were identified during a previous inspection were re-inspected and determined not to have changed. The aforementioned coating deficiencies were not repaired due to limited outage scheduling. Coating deficiencies are scheduled to be repaired this outage. The “B” sump was not accessible for inspection.

Intermediate Level

The coating system at the containment intermediate level appears to be generally in fair to good condition. Small (1 sq. in. size) to medium (5 sq. ft. size), localized coating defects were identified on components, pipes, internal structures, and concrete floors and walls. Coating defects include: mechanical damage, burnt coatings, flaking coating, delamination, and checking coatings. Checking coating was mainly identified on concrete walls. It appears the checking coating was applied over an existing coating. Documentation to

validate past recoating activities was not provided. The components exhibit general surface corrosion. No evidence of pitting corrosion was observed. Coating defects on steel and concrete were documented and prioritized for future monitoring and remediation. Previously identified coating deficiencies have not significantly changed.

Operating Level

The coating system at the operating level has not significantly changed since the previous inspection and appears to be generally in good condition. The interior containment liner was not inspected at this level due to installed insulation panels. Localized to randomly-distributed coating defects 1 sq. ft. to 5 sq. ft., were identified on components, pipes, internal structures, concrete floors and walls. Coating defects include: mechanical damage caused by equipment and material handling and relocation, burnt coatings caused by grinding and welding operation, flaking coating, and cracked/checking coatings. Checking coating was mainly identified on concrete walls. Exposed carbon steel substrate areas exhibit general light surface corrosion. No evidence of pitting corrosion was observed. Coating deficiencies were documented and prioritized for future monitoring and remediation.

Containment Dome

The interior containment dome is coated with inorganic zinc primer. Unlike other areas of the liner, the dome is not covered with insulation panels. The inorganic zinc primer on the dome was inspected remotely utilizing a high power flashlight. The inorganic zinc primer appears to be generally in fair to good condition. However, numerous localized areas of pinpoint rusting were identified and photographed. Pinpoint rusting is generally caused by insufficient uniform coating thickness applied to a substrate during initial application, or additionally resultant of the sacrificial deteriorating process of the zinc primer thus exposing substrate peaks. It does not appear that the condition of the coating has significantly changed from the last inspection cycle.

Discussion and Summary

UESI performed an inspection and assessment of the protective coatings applied to structures and components inside the GNPP primary containment. Specific emphasis was placed on coatings applied to accessible areas of the pressure boundary (liner). The objective of the assessment is to identify newly degraded coating and/or apparent unqualified coatings, quantify the extent of these conditions and make a comparative assessment with previous inspection data. Coating deficiencies identified during previous inspections were inspected and found to exhibit little or no significant changes.

Inspection data collected will assist site engineering in determining: (1) the effects of degraded coating on plant operation, to ensure that the ECCS and the safety-related CSS remain capable of performing their intended safety functions of safety related systems; and to mitigate corrosion of the containment liner and its integral structural and mechanical components.

Coating repair work was performed during G1R40 on fifteen (17) components located inside the primary containment building. The components are as follows: Penetrations 321, 322, 315, 311, 319 and 308; 9705 Valve Body; AFU29 Snubber Bracket; A&B Accumulators; SIU-130 Pipe Support; Liner moisture barrier area; "A" Sump; ACF05A Missile Shield and Support Columns 101, 113 and 114. Coating application and inspection were performed in accordance with Painting Application and Inspection Procedure, GC-76.11, Revision 01003, which was reviewed and approved by UESI.

Conclusions and Recommendations

As stated earlier, the overall coatings inside the primary containment are in fair to good condition. The coating assessment identified twenty-three (23) new areas of coating degradation that are recommended for repair to mitigate corrosion. No current coating conditions were observed that could impact structural integrity, plant operations, or safe shutdown. There is a significant amount of mechanical damage to the concrete floors throughout the containment building.

Coating Systems and Maintenance management should continue planning and scheduling coating repair work to be accomplished during future outages. The schedule should include areas of coating degradation identified during G1R40 as well as previous outages. An effectively executed repair plan will reduce radiation levels resulting from fixed contamination on exposed substrates, (both steel and concrete) mitigate progressive coating degradation by removing loose coating back to tightly adherent coating and making required safety-related coating repairs each RFO.

The site should continue to randomly remove insulation panels from the liner in the basement level as well as other areas to inspect the liner and concrete floor/liner transition area (moisture barrier). Pending inspection results, painters will have sufficient time to repair coating deficiencies identified without impacting the outage schedule. A database of degraded coating and areas repaired should be established to effectively monitor, document and track coating repairs completed each outage.

IWE Examination RFO G1R39 Fall 2015

The purpose of this examination was to ensure that the structural integrity of the ASME Class MC containment liner was maintained. Condition assessment of Class MC metallic liner was achieved by the performance of visual examinations of the accessible surfaces. During the Fall 2015 G1R39 IWE inspection the following items were identified:

Leak-Chase Channel Plugs

As part of the investigation into IN 2014-07, a first-time examination of the leak-chase channels was performed. In total, GNPP has 50 leak-chase channel plugs. During the examination, 19 of the 50 plugs were found to be covered by concrete and were therefore not accessible for examination. The remaining 31 leak chase channel plugs were found to be acceptable with no missing or damaged plugs noted.

Moisture Barrier

VT-3 of the moisture barrier between circumference 0- and 120-degrees azimuth was found unacceptable with heavy rusting in 8 spots. Extent of condition found medium to heavy surface rust and discoloration and coating failure extending under adjacent insulation panels. In total, 41.66 ft. were found to be degraded. Area was subsequently cleaned and VT-1 and ultrasonic examinations were performed following surface preparation with acceptable results. All ultrasonic measurements were well above the minimum required wall thickness of 0.300-in. with the most significant reduction in wall thickness exhibiting a remaining wall thickness of 0.387 in. Following examination, the area was recoated and VT-3 examination was performed with acceptable results.

During the examination, a portion of the containment liner plate was found to have a lack of complete coating. The area of concern was approximately 18-in. in length and 10-in. tall. The area was investigated by NDE and found to be acceptable. GNPP Engineering evaluated the condition and determined that this area had some lack of coating and had minor surface rusting, with no structural concerns. A VT-3 examination was performed after coating repairs were made to the containment liner. These areas were found to be acceptable with no reportable indications.

Moisture Barrier Technical Evaluation 2015

During the GNPP 2015 RFO, while performing the inspection of the containment liner moisture barrier in the containment building basement, degraded caulking and signs of discoloration on the concrete floor were observed in one area. The moisture barrier and associated sealed insulation were removed from this area (approximately 18 in. high X 7 ft. long) in order allow examination of the containment metal liner and concrete floor to liner moisture barrier region. Rust was observed on the carbon steel containment liner plate.

An area of moisture barrier degradation was discovered at the reactor building inside containment liner sealed insulation stainless steel facing to the reactor building floor. Subsequent inspections performed with the sealed insulation removed identified areas of liner corrosion requiring further examination and evaluation.

The Moisture Barrier (caulk) encompasses the entire length of the interface between the sealed insulation over the metal liner and the containment basement concrete floor. This configuration makes the metal liner inaccessible. The purpose of this document is to evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or could result in degradation to such inaccessible areas.

The entire GNPP Moisture Barrier Examinations were performed in 2012 and found acceptable. The 2012 examinations covered the full length of the containment moisture barrier.

GNPP has identified one suspect area (Penetration 29) and examinations were performed in this area every period (every 3 to 4 years) since 2005. This suspect area is located at Penetration 29 and the basement floor to the sealed insulation lower panel at 35 Ft West of Penetration 29 and 30 ft. east of Penetration 29. The caulk and lower panels of sealed insulation are removed and associated VT-3 (visual examination), VT- 1 (visual examination) and UT (ultrasonic thickness) examinations performed. Upon acceptable completion, the sealed insulation was re-installed, the Moisture Barrier re-installed, and VT-3 re-examined. The associated examinations were acceptable.

The most recent Moisture Barrier Examinations were performed in 2014 and 2015. The result of the 2015 RFO Moisture Barrier examination was acceptable with the exception of one location. The 2015 RFO examinations covered all moisture barriers except for the Penetration 29 area. The 2014 RFO examination was associated with Penetration 29 area and all examinations associated with this area were found acceptable.

In the past, GNPP has performed examinations, which include the following:

During the 1999 RFO, a one-third of the Moisture Barrier was examined and one area was identified that the caulking design detail did not conform to the design specifications. As a result of this discovery, the inspection scope was increased to include the entire circumference of the Moisture Barrier. The caulking was found to be continuous with no visible gaps or discontinuities. As a result of this non-conformance caulk detail issue, the sealed insulation was removed in two areas (one on the north side and one on the west side). Six to eight ft. of the metal liner was exposed and visually examined. On the north end, evidence of minor surface corrosion was visible. UT Thickness readings were taken and ranged from 0.346 in. to 0.404 in. on the north side and 0.388 in. to 0.404 in. on the west side. The nominal thickness of the Containment Liner is 0.375 in. The minimum required thickness was determined by engineering analysis to be 0.281 in. The area was recoated and the sealed insulation re-installed. The Moisture Barrier was replaced and baseline examination performed and acceptable.

During the 2002 RFO, approximately 70 linear ft. of Containment Liner was exposed for visual inspection on the south-east side of Containment. In four different areas, the concrete floor was excavated to a depth of 1 ½ in. below the floor level over a length of 12. Ultrasonic Thickness measurements were taken in these areas. The minimum thickness readings for these locations were 0.402 in., 0.412 in., 0.395 in. and 0.402 in. Visual evidence of superficial surface corrosion was present at various areas along the exposed portion of the liner. As a result, the liner was cleaned and restored with new coating of zinc-rich paint. Insulation was restored and the Moisture Barrier installed. In response to License Renewal RAIs, the containment liner shall be restored if the liner thickness is at or below 0.300 in. This new value provides margin with respect to the minimum wall thickness requirement of 0.281 in. The nominal wall thickness of the containment liner is 0.375 in.

During the 2006 RFO, a 20-foot section of the Metal Liner was exposed in the northwest quadrant of containment for a License Renewal Examination. Visual

and UT thickness examinations were performed and acceptable. The lowest thickness reading was 0.382 in. and was above the License Renewal commitment of 0.300 in. threshold for restoring the liner.

During the 2015 RFO, a containment liner moisture barrier at the 235 ft.-8 in. elevation was examined (from 0 degrees to 120 degrees azimuth under ISI Summary Number 1900932), and only one localized area was identified for follow-up inspection in the northeast section of containment. The degraded area, located at approximately 50 degrees azimuth from North, 6 ft. in length, was noted with rust stains running out from under the caulking/insulation and onto the floor. The degraded Moisture Barrier and associated sealed insulation were removed from this area (approximately 18 in. high by 7 ft. long) in order to allow examination of the containment metal liner. Rust and pitting was observed on the carbon steel containment liner plate in a band within two to three in. above the concrete basement floor. Light residual dry boric acid was identified. Ultrasonic (UT) thickness measurements of the liner plate show the actual thickness of the 3/8 in. plate to be 0.405 in. The mechanical measurement of the depth of the worst case pitting was 1/16 or 0.063 in, resulting in a measured degraded thickness of 0.342 in. The minimum allowable thickness, based on License Renewal commitment is 0.300 in.

Additional insulation panels were removed to identify extent of condition. A total of 500 in. of metal liner was exposed. The two to three in. band area was conditioned, cleaned, boric acid residue removed, visually examined and acceptable. Ultrasonic thickness readings were also performed on the expanded area, the lowest reading obtained was 0.387 in. This reading is acceptable and above license renewal commitment of 0.300 in. The area was recoated (painted) and a pre-service visual examination was performed and acceptable. The area was recovered with sealed insulation and the moisture barrier reapplied. A pre-service visual examination was performed on the new moisture barrier and was found acceptable.

It is postulated that the degraded section of Moisture Barrier (caulk) allowed standing floor water to penetrate into the insulation. The insulation became wet by wicking that resulted in a band of corrosion within two to three in. along the floor. The floor water was caused by normal outage and refueling activities such as Reactor Cavity Leakage. As a result of cavity leakage, Penetration 29 area has been identified as a Section XI Code Suspect Area since 2005, and is located roughly 180 degrees azimuth from North (0 degrees azimuth). This suspect area experienced continuous wetting and is located along the basement floor to the sealed insulation and moisture barrier at Penetration 29, 35 ft. West of Penetration 29 and 30 ft. east of Penetration 29. The lower panels of sealed insulation are removed and associated VT-3, VT-1 and UT thickness examinations performed. Upon acceptable completion, the sealed insulation and moisture barrier was re-installed, VT-3 re-examined and acceptable. The 2015 RFO area was between 10 to 70 degrees azimuth and away from the Penetration 29 suspect area located at 180 degrees azimuth.

In summary:

a. The visual inspection of the containment liner Moisture Barrier at the 235

ft.-8 in. elevation was performed from 0 degrees to 120 degrees azimuth and only one localized area was identified for follow-up inspection in the northeast section of containment. The area was noted with rust stains running out from under the insulation and onto the floor. The moisture barrier insulation was removed and some rust and pitting were identified on the carbon steel containment liner plate in a band within two to three in. above the floor. The moisture barrier (caulk) degraded in a section that allowed containment standing floor water to penetrate the degraded moisture barrier and wet the insulation by a wicking action. UT thickness measurements of the base material show the actual thickness of the 3/8" plate to be 0.405 in. The mechanical measurement of the depth of the worst case pitting was 1/16 or 0.063 in, resulting in a measured degraded thickness of 0.342 in. (the minimum allowable thickness is 0.300 in.).

b. Besides the original 7 ft. section of sealed insulation panel removed, as an extent of condition, a total of five additional 7 ft. sections of sealed insulation panels have been removed for inspection of the adjacent areas. The total length of exposed metal liner equates to 500 in. Based on the VT-3 visual inspections and mechanical measurements, the original area of degradation was the most severe and had the most material loss, therefore, bounding all other inspected areas.

c. The liner-rusted surface was in a band within two to three in. above the floor. This area was prepped for a VT-1 and UT inspection in order to determine remaining wall thickness of the containment liner. Associated visual examinations and thickness readings were acceptable. The liner areas were cleaned, recoated, VT-3 examined and found acceptable. The Moisture Barrier was re-installed, examined and found acceptable.

d. Currently, one suspect area (Penetration 29) has been identified since 2005 and is located roughly 180 degrees azimuth from North. This suspect area is at the basement floor to the sealed insulation at 35 ft. west of Penetration 29, 30 ft. east of Penetration 29, and Penetration 29. The caulk and lower panels of sealed insulation are removed and associated VT-3, VT-1 and UT thickness examinations are performed. Upon acceptable completion of examinations, the sealed insulation is re-installed, the Moisture Barrier reinstalled and VT-3 re-examined. This area was on an increased inspection frequency as required by ASME Section XI Code, 2004 Edition, no Addenda. The increased frequency of examination on this area is every period. The degraded areas identified during the 2015 RFO are not associated with this area.

e. During the 2015 and 2014 RFOs, the Moisture Barrier was 100% examined. The only unacceptable condition was identified during the 2015 RFO as identified above. Based on these facts, it can be concluded that Moisture Barrier examinations are successful at identifying unacceptable degraded conditions. There is currently no evidence to indicate that other acceptable areas would have equal or greater material loss.

f. The most likely cause for the observed degradation is the exposure to

borated water from leakage of the reactor cavity during refueling activities. This degradation mechanism is a long-term issue, potentially corroding the liner over the life of the plant only when the Moisture Barrier does not perform its intended function. Refueling activities have been concluded for the 2015 RFO and the cavity has been drained. Based on this, it is not expected that the degradation will propagate. Recoating the metal liner and re-caulking of the Moisture Barrier have stopped the identified degradation of the metal liner.

IWE Examinations RFO G1R40 Spring 2017

Containment Vessel Dome Liner

VT-3 on containment dome revealed indications of staining (discoloration, general rust and oxidation) identified on dome surface. Conditions identified were assessed in accordance with GNPP visual examination procedure acceptance criteria. Staining and general rust indications were compared to indications recorded during the 2015 RFO. Additional areas of general rust and staining identified during the 2018 RFO were compared against the 2015 examination results and were found to be acceptable.

Mechanical Penetration 321A – Steam Generator Blowdown

VT-3 visual examination on mechanical penetration 321A found minor corrosion on the penetration to liner weld. Light chipping of the coating was observed throughout the penetration. Wear of coating to bare metal on face of penetration appeared to be due to insulation. No material loss due to general corrosion was observed.

Mechanical Penetration 412 – Main Steam from B Steam Generator

VT-3 visual examination on mechanical penetration 412 found light rust on welds with no metal loss.

IWL Examinations Fall 2010 and Fall 2015

The purpose of these examinations was to ensure that the structural integrity of the ASME Class CC reinforced concrete was maintained. Condition assessment of Class CC reinforced concrete was achieved by the performance of visual examinations of the accessible surfaces. During the 2010 and 2015 IWL inspections the following items were identified and are detailed in Table 3.6.6-1 below:

Table 3.6.6-1 2010 and 2015 Concrete Indications

Indication No.	Building	Azimuth	Elevation	Description/Remarks	2010 Engineering Remarks	Acceptable	2015 Engineering Remarks	Acceptable
1	Dome	61	28	Hole 0.600" x 1.365" deep	No change from previous test results	Yes	No change from previous test results	Yes
2	Dome	63	29	Crack on "B" SG Patch = 0.115"	No change from previous test results	Yes	No change from previous test results	Yes
3	Dome	65	29	Three cracks on "B" S/G Patch = 0.055", 0.050" & 0.115, same as Indication No. 2	No change from previous test results	Yes	No change from previous test results	Yes
4	Dome	65	30	Crack on "B" SG Patch continuation of Indication 3, 0.055" - 0.060"	No change from previous test results	Yes	No change from previous test results	Yes
5	Dome	65	31	Hole 0.600" rounded x 1.2" deep	No change from previous test results	Yes	No change from previous test results	Yes
6	Dome	71	37	Edge of patch cut, not filled = 1" triangle not filled x 1/2" deep	No change from previous test results	Yes	No change from previous test results	Yes
7	Dome	73	29	Air pocket 0.720" x 1.165" x 0.465" deep	No change from previous test results	Yes	No change from previous test results	Yes
8	Dome	73	30	Air pocket 0.450" x 0.165" deep and 0.320" x 0.255" deep	No change from previous test results	Yes	No change from previous test results	Yes
9	Dome	77	38	1" diameter rebar cut at surface	No change from previous test results	Yes	No change from previous test results	Yes
10	Dome	101	14	Rough surface at edge of patch	No change from previous test results	Yes	No change from previous test results	Yes

Table 3.6.6-1 2010 and 2015 Concrete Indications

Indication No.	Building	Azimuth	Elevation	Description/Remarks	2010 Engineering Remarks	Acceptable	2015 Engineering Remarks	Acceptable
11	Dome	105	15	Bottom edge of S/G Patch seam 0.030" to 0.090" wide	No change from previous test results	Yes	No change from previous test results	Yes
12	Dome	109	17	Bottom edge of S/G Patch has a 3/32" offset and 0.025" separation	No change from previous test results	Yes	No change from previous test results	Yes
13	Dome	131	38	Original pore seam 0.035" to 0.050" wide, also 0.275" x 0.900" x 0.225" deep void	No change from previous test results	Yes	No change from previous test results	Yes
14	Dome	163	28	Four air pockets, 0.700" x 1.1" x 0.250" deep, 0.600" - 0.250" and 0.360" rounded x 2.9" deep	No change from previous test results	Yes	No change from previous test results	Yes
15	Dome	163	38	Original seam with some eroded edges 0.060" to 0.600" wide, max. depth is 0.365"	No change from previous test results	Yes	No change from previous test results	Yes
16	Dome	185	14	Shrinkage type indication, 0.110" wide by 2.320" long with minimal depth	No change from previous test results	Yes	No change from previous test results	Yes
17	Dome	211	32	Plug patch tear 0.090" wide	No change from previous test results	Yes	No change from previous test results	Yes
18	Dome	211	35	Plug patch separation at top of plug = 0.070"	No change from previous test results	Yes	No change from previous test results	Yes

Table 3.6.6-1 2010 and 2015 Concrete Indications

Indication No.	Building	Azimuth	Elevation	Description/Remarks	2010 Engineering Remarks	Acceptable	2015 Engineering Remarks	Acceptable
19	Dome	229	8	Shrinkage type indication 0.070" wide x 2.65" long	No change from previous test results	Yes	No change from previous test results	Yes
20	Dome	233	9	Pop out at seam 1.56" x 2.20" x 0.675" deep	Pop-out filled	Yes	No change from previous test results	Yes
21	Dome	241	32	Void in plug patch 0.400" rounded x 0.400" deep	No change from previous test results	Yes	No change from previous test results	Yes
22	Dome	277	39	Eroded seam edge 0.370" to 0.400" wide, maximum depth is 0.280"	No change from previous test results	Yes	No change from previous test results	Yes
23	Dome	283	39	Eroded seam edge 0.280" to 0.600" wide x 0.300" deep max	No change from previous test results	Yes	No change from previous test results	Yes
24	Dome	301	21	Crack in excessive material on lower section of the "A" patch = 0.100" widest	No change from previous test results	Yes	No change from previous test results	Yes
25	Dome	301	22	Crack in excessive material same as Indication 24 = 0.060" wide in this area	No change from previous test results	Yes	No change from previous test results	Yes
26	Dome	357	27	Void at edge of "A" S/G Patch 0.700" x 1.3" long maximum depth is 0.400"	No change from previous test results	Yes	No change from previous test results	Yes
27	Dome	0 to 360		Grout at trunion is dis-bonded 360 degrees (118" circumference)	New indication 2010	Yes	No change from previous test results	Yes

Table 3.6.6-1 2010 and 2015 Concrete Indications

Indication No.	Building	Azimuth	Elevation	Description/Remarks	2010 Engineering Remarks	Acceptable	2015 Engineering Remarks	Acceptable
28	Dome	90 to 170		Segregation/wood embedded into concrete	New indication 2010	Yes	No change from previous test results	Yes
29	Dome	125		Efflorescence 4' x 2.5' area	New indication 2010	Yes	No change from previous test results	Yes
30	Dome	160		Cosmetic S/G Patch breakaway on "B"	New indication 2010	Yes	No change from previous test results	Yes
31	Dome	26 to 320	351	Cosmetic S/G Patch breakaway at original pour seam located at tendon cans 46 through 48, 52 through 54, 101, 111 and 128	CR-2010-003972 - what appeared to be spalling was actually the cosmetic S/G Patch breaking away at seam	Yes	No change from previous test results	Yes
32	Dome	85	38	The edge of "B" S/G Patch has a saw mark 0.25" wide and a void at the edge of patch 0.44" wide	New indication 2010	Yes	No change from previous test results	Yes
33	Dome	87		The edge of "B" S/G Patch has a small void at the edge of the patch 0.188" wide	New indication 2010	Yes	No change from previous test results	Yes
34	Dome	0 to 360	0 to 3	Intermittent cracks / linear indications noted under paint less than 0.020" / efflorescence	New indication 2010	Yes	No change from previous test results	Yes

Table 3.6.6-1 2010 and 2015 Concrete Indications

Indication No.	Building	Azimuth	Elevation	Description/Remarks	2010 Engineering Remarks	Acceptable	2015 Engineering Remarks	Acceptable
35	Dome	0 to 360	347 to the Top	Seam cracks noted from the original pour ranging from 0.016" to 0.030"	New indication 2010	Yes	No change from previous test results	Yes
36	Dome	250 to 285	9	Cosmetic S/G Patch at seam is dis-bonded	New indication 2010	Yes	No change from previous test results	Yes
37	Dome	233	9	Void 2" x 1" 3/8" depth	New indication 2010	Yes	No change from previous test results	Yes
38	Dome	354 to 140	347	Cosmetic patch at seam is dis-bonded above tendon cans 69 through 74, 77 through 78, 106 through 107, 126 through 131, and 133 through 135	New indication 2010	Yes	No change from previous test results	Yes
39	Tendon Cans		0	Behind the tendon cans there are divots which do not look service induced: may have been removed to make room for the bearing plates located at tendon cans 1 through 10, 15 through 20, 35, 50, 51, 53, 133, 135, 144 through 148, 150 through 152, 156 and 158	New indication 2010	Yes	No change from previous test results	Yes

Table 3.6.6-1 2010 and 2015 Concrete Indications

Indication No.	Building	Azimuth	Elevation	Description/Remarks	2010 Engineering Remarks	Acceptable	2015 Engineering Remarks	Acceptable
40	Dome	283 to 305	0	Erosion / loss of material from base of tendon cans 38 through 47 (20' x 3' x 1" depth)	New indication 2010	Yes	Repaired with grout	Yes
41	Dome	153	347	Efflorescence 3" x 4' area	New indication 2010	Yes	No change from previous test results	Yes
42	Dome	345	363	Erosion noted on permanent mounted plate / footer for scaffolding - depth varies from 0.5" to 4" - 34" length x 24" width	New indication 2010	Yes	No change from previous test results	Yes
43	Dome	314		"A" S/G Patch "crack-like" indication	New indication 2010	Yes	No change from previous test results	Yes
44	Dome	42,71,222	0	CR-2010-004110 - At the edge of "A" S/G Patch - there is not a smooth transition, which gives the appearance of a large "crack-like" indication on the dome itself. There is some fine cracking on the S/G Patch, but not recordable.	New Indication 2010	Yes	No change from previous test results	Yes

Table 3.6.6-1 2010 and 2015 Concrete Indications								
Indication No.	Building	Azimuth	Elevation	Description/Remarks	2010 Engineering Remarks	Acceptable	2015 Engineering Remarks	Acceptable
45	Dome	197 to 200	0	Saturated pig placed between the containment wall and tendon cans causing some erosion to the wall and serving no positive action	New indication 2010	Yes	No change from previous test results - Pig has been removed	Yes
46	Dome	0 to 360	0	Patches excessive material adjacent to the bearing plate	New indication 2010	Yes	No change from previous test results	Yes
47	Dome	0 to 360	0	10% linear indications around adjacent areas of bearing plates - indications less than 0.010"	New indication 2010	Yes	No change from previous test results	Yes
48	Tendon Can		0	Grease present at the base / top of tendon can no. 130 - possible reside from filling can or gasket leak	New indication 2010	Yes	No change from previous test results - no grease observed	Yes
49	Tendon Can		0	Bolt length on the top of tendon cans no. 111, 117, 118 was insufficient - the nut is not fully engaged	New indication 2010	Yes	No change from previous test results - no grease observed	Yes

Table 3.6.6-1 2010 and 2015 Concrete Indications

50	Dome	0 to 360	342	Joint sealer missing and /or deteriorated intermittently 360 degrees around	N/A -Indication found in 2015	N/A	Registered Professional Engineer (RPE) dispositioned with VT-1C and found acceptable.	Yes
51	Dome	290 to 315	342	Grouted areas between tendon cans are cracking and breaking away.	N/A -Indication found in 2015	N/A	RPE dispositioned with VT-1C and found to be acceptable.	Yes
52	Dome	290 to 315	342	Tendon can gasket is degraded on cans 46, 53, and 57	N/A -Indication found in 2015	N/A	RPE dispositioned with VT-1C and found acceptable.	Yes
53	Dome	SG Patch Areas	N/A	Steam Generator patches have intermittent areas along the seams where the blended area is breaking away	N/A -Indication found in 2015	N/A	RPE dispositioned with VT-1C. Indications examined by RPE with recommended repair completed on 10/14/2016	Yes

Tendon Surveillance Assessment October 2010

This report details the 2010 GNPP containment structure post-tensioning system tendon surveillance performed by Precision Surveillance Corporation (PSC). The surveillance program is a systematic means of assessing the quality and structural performance of the post-tensioning system.

The tendon surveillance program consists of a periodic inspection of the condition of a selected group of tendons. This program provides confidence in the condition and functional capability of the system, and an opportunity for timely corrective measures if adverse conditions are detected. The 2010 tendon surveillance at GNPP began on June 14th, 2010, and was completed on June 30th, 2010. This surveillance period consisted of a Physical Inspection of the post-tensioning system. Physical tendon surveillance consists of: sheathing filler inspection and testing, inspection for water, anchorage inspection, concrete inspection around tendons, force monitoring, inspection and tensile testing of removed wire samples and replacement of grease after completion of all inspections. All procedures completed during this surveillance were performed on the top ends of the surveillance tendons, due to the inaccessibility of the couplers on the bottom ends.

This examination was performed in accordance with the requirements of the ASME BPV Code, Section XI, 2004 Edition, and the applicable amendments, as specified in 10 CFR 50.55a, Codes and Standards.

A review of this surveillance was conducted per IWL-3221, Un-bonded Post-Tensioning Systems, and is outlined below.

IWL-3221 – Acceptance by Examination

IWL-3221.1 – Tendon Force. Tendon forces are acceptable if:

a) The average of all measured tendon forces, including those measured in IWL-3221.1(b)(2), for each type of tendon is equal to or greater than the minimum required pre-stress specified at the anchorage for that type of tendon.

Results: The as-found forces for each inspected tendon were above the corresponding required minimum design force values provided by the utility.

b) The measured force in each individual tendon is not less than 95% of the predicted force unless the following conditions are satisfied:

- The measured force in not more than one tendon is between 90% and 95% of the predicted force;
- The measured forces in two tendons located adjacent to the tendon in IWL-3221.1(b)(1) are not less than 95% of the predicted forces; and
- The measured forces in all the remaining sample tendons are not less than 95% of the predicted force.

Results: All of the tendon liftoffs were found within the acceptable levels required by GNPP's Tendon Surveillance Program, STP-O-27.2, of minimum 636,000 pounds force (636 kips) and maximum of 750 kips.

IWL-3221.2 – Tendon Wire or Strand Samples. The condition of wire or strand samples is acceptable if:

a) Samples are free of physical damage.

Results: All of the tendon wire test samples were free of physical damage.

b) Sample ultimate tensile strength and elongation are not less than minimum specified values.

Results: All of the tendon test wire samples had acceptable results for ultimate tensile stress (≥ 240 thousand pounds per square inch) (ksi) and elongation (≥ 4%).

Tendon Surveillance Assessment October 2011

During the June 2010 performance of STP-O-27.2 "Tendon Surveillance Program," part of the procedure – the 106% overstress test - was not performed by the test vendor. The required section of the procedure was marked N/A, and then inappropriately deleted using the Step Delete process during WO closeout. The issue was discovered while researching tendon surveillance history in support of an upcoming modification.

GNPP's containment tendon design is one-of-its-kind in the industry, and the overstress test is a unique requirement based on accident analyses assumptions. The overstress test is not required by ASME code or the applicable Regulatory Guide. It has been part of the required GNPP surveillance since the original 1969 Technical Specifications, and is currently on a 5 year frequency.

Following the omission of the 106% overstress test portion of the Tendon Surveillance Test, a major procedure revision was completed on STP-O-27.2 (formerly PT-27.2). This Tendon Surveillance Test was re-performed satisfactorily in the spring of 2011.

If the surveillance had not been fully completed prior to its late end date, this condition would have resulted in containment being declared inoperable per TS SR 3.6.1.2. The last completed surveillance was approved on 10/04/2005. Data collected in the incomplete 2010 surveillance did not identify a degraded condition, and review of test reports did not indicate any reason to doubt that the overstress test results would have been satisfactory. TS SR 3.0.2 is applicable, and allows 1.25 times the specified frequency for completion. Therefore, completion of the required tests prior to 01/02/2012 fulfilled the surveillance requirement. As mentioned above, the surveillance was completed during the spring of 2011.

This report details the 2011 GNPP containment structure post tensioning system tendon surveillance. The surveillance program is a systematic means of assessing the quality and structural performance of the post tensioning system.

This examination was performed in accordance with the requirements of the ASME BPV Code, Section XI, 2004 Edition, and the applicable amendments, as specified in 10 CFR 50.55a, Codes and Standards.

A review of this surveillance was conducted per IWL-3221, Un-bonded Post-Tensioning Systems, and is outlined below.

Fourteen (14) tendons were initially selected for the 2011 tendon surveillance at Ginna. Per page 5 of STP-O-27.2, seven (7) tendons were added to the inspection scope for data collection.

Tendon V-75 was previously identified as having 24 missing wires. Because of this, Tendon V-75 underwent a visual inspection and grease analysis for information only. An inspection for effective wires was performed, and no change was noted since the last visual inspection of V-75. Tendon V-75 was found to have 24 missing wires, as previously reported. Tendons V-74 and V-76, located on either side of V-75, were fully tested.

The results of this investigation are summarized as follows:

1. The sheathing filler (grease) samples were tested and found to have acceptable levels of water-soluble ions (Chlorides, Nitrates and Sulfides). The moisture contents were all below the acceptable limit of 10% water by weight. All but four of the grease samples tested had neutralization numbers greater than zero and are acceptable. Four of the samples resulted in neutralization numbers below detection limits. Additional acid tests were conducted on these samples to verify the low numbers. These acid tests produced results below detection limits as well, indicating a near neutral condition.

2. No tendons showed presence of water during the removal of the grease cap, during anchorage inspection or at any other point during inspections.

3. The bearing plate of tendon V-91 was found with a Level 3 corrosion level. The bearing plate was prepped and re-coated. This work was completed on 4/13/2011. Acceptable corrosion levels were found on all components of other tendon ends and no cracks were found on any anchorage components.

4. No missing or protruding button-heads were identified during anchorage inspections that were not previously reported.

5. A detailed visual inspection was performed on the 24” of concrete surrounding the bearing plate of each tendon end inspected. No recordable indications were noted during these examinations.

6. The hydraulic jack used for tendon liftoffs was calibrated and found to be within an acceptable variation of +/- 1.5% as calculated using the maximum calibration force.

7. All of the tendons monitored for forces this inspection period were found to have forces within the acceptable values, as specified by Ginna's surveillance procedure. Each tendon liftoff force was greater than the calculated Predicted Force for that tendon.

8. An overstress test was performed on each physical surveillance tendon. The overstress test applies an additional 6% jacking stress over the liftoff pressure to verify the ability of the tendon to sustain the added stress applied during the design basis accident. The elongation measurement taken coincident with the overstress test is used to investigate the state of the embedded rock anchors at the bottom of the tendon. The elongation is used to confirm no instant creep resulted from the overstress loading. Each tendon is stressed to 106% of their recorded tendon liftoff force or 848kips, whichever is less during the surveillance. Upon completion of the overstress test, the tendon's button-heads are visually examined for any evidence of damage. No damage to the button-heads was recorded for any tendon. No damage to any anchorage components was recorded as a result of the overstress tests.

9. All test wires removed from tendons were found to have acceptable corrosion levels. All tendon test wire samples had acceptable diameter, yield stress, ultimate stress and elongation results.

10. All tendons were resealed and re-greased.

11. A comparison of "As-found" force levels to the forces measured during Ginna's 30th year tendon surveillance was performed and no abnormal average force difference was observed.

Based on the data gathered during the GNPP 40TH Year In-Service Inspection on the containment structure post tensioning system, the conclusion is reached that no abnormal degradation of the post tensioning system has occurred.

Tendon Surveillance Assessment October 2016

This report details the 2016 GNPP containment structure post tensioning system tendon surveillance performed by PSC. The surveillance program is a systematic means of assessing the quality and structural performance of the post- tensioning system.

The tendon surveillance program consists of a periodic inspection of the condition of a selected group of tendons. This program provides confidence in the condition and functional capability of the system, and an opportunity for timely corrective measures if adverse conditions are detected. The 2016 tendon surveillance at GNPP began in March 2016 and was completed in April 2016. This surveillance period consisted of a Physical Inspection of the post-tensioning system. Physical tendon surveillance consists of: sheathing filler inspection and testing, inspection for water, anchorage inspection, concrete inspection around tendons, force monitoring, inspection and tensile testing of removed wire samples, and replacement of grease after completion of all inspections. All procedures completed during this surveillance were performed on the top ends of the surveillance tendons.

This examination was performed in accordance with the requirements of the ASME BPV Code, Section XI, 2004 Edition, and the applicable amendments, as specified in 10 CFR 50.55a, Codes and Standards

A review of this surveillance was conducted per IWL-3221, Un-bonded Post-Tensioning Systems, and is outlined below.

IWL-3221 – Acceptance by Examination

IWL-3221.1 – Tendon Force and Elongation. Tendon forces and elongation are acceptable if the following conditions are met:

a) The average of all measured tendon forces, including those measured in IWL-3221.1(b)(2), for each type of tendon, is equal to or greater than the minimum required pre-stress specified at the anchorage for that type of tendon.

Results: The average of all measured tendon forces was above the minimum required pre-stress of 636 kips, as described by GNPP's Tendon Surveillance Program STP-O-27.2.

b) The measured force in each individual tendon is not less than 95% of the predicted force unless the following conditions are satisfied.

- The measured force in not more than one tendon is between 90% and 95% of the predicted force;
- The measured forces in two tendons located adjacent to the tendon in IWL-3221.1(b)(1) are not less than 95% of the predicted forces; and
- For tendons requiring augmented examination in accordance with Table IWL-2521, Item L2-10, the measured forces in two like tendons located nearest to, but on opposite sides of, the tendon described in IWL 3221-1(b)(1) are not less than 95% of the predicted forces.
- The measured forces in all the remaining sample tendons are not less than 95% of the predicted force.

Results: All the tendon liftoffs were found to be within the acceptable levels required by GNPP's Tendon Surveillance Program STP-O-27.2 of a minimum of 636 kips and maximum of 750 kips. Each tendon was found to have a liftoff force greater than 95% of the tendon Predicted Force, as required by IWL 3221.1(b).

c) The pre-stressing forces for each type of tendon measured in IWL-3221.1(a) and (b), and the measurement from previous examination, indicate a pre-stress loss such that predicted forces meet the minimum design pre-stress forces at the next scheduled examination.

Results: This Code requirement analysis was performed by GNPP as part of their owner's acceptance evaluation.

d) The measured tendon elongation varies from the last measurement, adjusted for effective wires or strand, by less than 10%.

Results: Due to plant configuration vertical tendons are not de-tensioned and accordingly measurement is not performed relative to original installation.

IWL-3221.2 – Tendon Wire or Strand Samples. The condition of wire or strand samples is acceptable if:

a) Samples are free of physical damage;

Results: All of the tendon wire test samples were free of physical damage.

b) Sample ultimate tensile strength and elongation are not less than minimum specified values.

Results: All of the tendon test wire samples had acceptable results for ultimate tensile stress (≥ 240 ksi) and elongation (≥ 4%).

IWL-3221.3 – Tendon Anchorage Areas. The condition of tendon anchorage areas is acceptable if:

a) There is no evidence of cracking in anchor heads, shims, or bearing plates;

Results: Detailed inspections did not reveal any cracks in the anchorage components for any inspected tendon end.

b) There is no evidence of active corrosion;

Results: No tendon end inspected revealed active corrosion on the anchorage components.

c) Broken or unseated wires, broken strands, and detached button-heads were documented and accepted during a pre-service examination or during a previous inservice examination;

Results: No missing or protruding wires/button-heads were discovered during the examinations of surveillance tendon ends that were not previously reported.

d) Cracks in the concrete adjacent to the bearing plates do not exceed 0.01 in. (3mm) in width;

Results: No cracks exceeding 0.010 in. were detected in the 24 in. of concrete adjacent to the bearing plates of the tendon ends inspected.

e) There is no evidence of free water.

Results: No water was detected on any of the inspected tendons at any point during inspections except condensation drops on V-87.

IWL-3221.4 – Corrosion Protection Medium. Corrosion protection medium is acceptable when the reserve alkalinity, water content and soluble ion concentrations of all samples are within the limits specified in Table IWL-2525-1. The absolute difference between the amount removed and the amount replaced shall not exceed 10% of the tendon net duct volume.

Results: All sheathing filler (grease) samples were tested and found to have acceptable levels of water-soluble ions (Chlorides, Nitrates and Sulfides). Water content values were below 10% by weight and acceptable for all samples tested. All but one of the grease samples tested had neutralization numbers greater than zero, and are found to be acceptable. The one samples had a result in neutralization number below detection limits. An additional acid test was conducted on this sample to verify the low number. This acid test produced a result below detection limits as well, indicating a near neutral condition.

IWL-2525.2(b) states, "Free water samples shall be analyzed to determine pH."

Results: The amount of condensation identified in V87 (drops) was insufficient to collect and test for pH.

Based upon the evaluation of the ISI results for the GNPP 45th Year Containment Building Tendon Surveillance reported herein, PSC concludes that the containment structure has experienced no abnormal degradation of the post-tensioning system.

2014 and 2017 Appendix J Containment Visual Structural Inspections

The general visual structural inspection of the accessible interior and exterior surfaces of the containment structure was performed with existing liner plate insulation panels in place. The inspection is designed to uncover any evidence of structural deterioration which could affect either the containment structural integrity, its leak tightness, or the performance of the ILRT. The 2014 and 2017 performance of the GNPP Containment Visual Structural inspections were found to be satisfactory with no containment structural issues identified.

3.7 Containment Modifications

No major containment modifications have been performed since the last ILRT in 2011.

3.8 License Renewal Aging Management

By letter dated July 30, 2002, EGC requested renewal of the operating licenses issued in Section 104b (Operating License No. DPR-18) of the Atomic Energy Act of 1954, as amended, for GNPP for a period of 20 years beyond the current license expiration dates of midnight September 18, 2009 (Reference 21). The following programs, which are part of the supporting basis of this LAR, are also Aging Management Programs at GNPP.

3.8.1 Aging Management Programs

Appendix J Program

The 10 CFR Part 50, Appendix J Program consists of monitoring of leakage rates through containment liner/welds, penetrations, fittings and access openings to detect degradation of the pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. This Program is implemented in accordance with Option B (performance-based leak rate testing) of 10 CFR 50, Appendix J; RG 1.163; and NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J." This Program is consistent with the corresponding program described in NUREG-1801 (Reference 22).

ASME Section XI, Subsection IWE Program

The ASME Section XI, Subsection IWE Program consists of periodic inspection of steel containment components for signs of degradation, assessment of damage, and corrective actions. This Program is conducted in accordance with ASME Section XI, Subsection IWE, and in accordance with 10 CFR 50.55a. The ASME Section XI, Subsection IWE Program is consistent with the corresponding program described in NUREG-1801 (Reference 22).

ASME Section XI, Subsection IWL Program

The ASME Section XI, Subsection IWL Program is credited for the aging management of accessible and inaccessible pressure retaining Primary Containment concrete. The GNPP containment structure employs the use of post-tensioned pre-stressing tendons. Therefore, the ASME Section XI, Subsection IWL rules regarding post-tensioning systems are applicable. This Program is conducted in accordance with the ASME Section XI, Subsection IWL and in accordance with 10 CFR 50.55a. The ASME Section XI, Subsection IWL Program is consistent with the corresponding program described in NUREG-1801 (Reference 22).

Protective Coating Monitoring and Maintenance Program

Proper maintenance of protective coatings inside containment (described as Service Level 1 in NRC Regulatory Guide 1.54, Rev. 1) is essential to ensure operability of post-accident safety systems that rely on water recirculated through the containment emergency Sump "B." GNPP maintains protective coatings inside containment in accordance with our program as described in our December 1, 1998 response to Generic Letter 98-04, to ensure that paint chips or flakes do not dislodge in a post-accident environment and cause unacceptable sump blockage.

The Protective Coatings Monitoring and Maintenance Program inside containment, although not developed in accordance with Regulatory Guide 1.54 and ASTM D5163-96, is consistent with the NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.S8, "Protective Coatings Monitoring and Maintenance". This program was described in detail in the December 1,

1998 response to Generic Letter 98-04, and was accepted by the NRC in their letter of November 19, 1999. Although consistent with NUREG-1801 it is not considered an aging management program, but is described to demonstrate compliance with the resolution of GSI-191.

3.9 NRC-SER Limitations and Conditions

3.9.1 Limitations and Conditions Applicable to NEI 94-01, Revision 2-A

The NRC staff found that the use of NEI TR 94-01, Revision 2, was acceptable for referencing by licensees proposing to amend their TS to permanently extend the ILRT surveillance interval to 15 years, provided the following conditions, as listed in Table 3.9.1-1, were satisfied:

Table 3.9.1-1 – NEI 94-01, Revision 2-A, Limitations and Conditions

Limitation/Condition (From Section 4.0 of SE)	GNPP 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. (Refer to SE Section 3.1.1.1.)	GNPP will utilize the definition in NEI 94-01, Revision 2-A, Section 5.0
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 Sections 3.5.3 and 3.5.4 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 3.5.3 of this submittal.
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. No containment or containment isolation system modifications were required at GNPP to comply with the NRC Orders for FLEX. Reference Section 3.7 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.)	GNPP will follow the requirements of NEI 94-01, Revision 2-A, Section 9.1. In accordance with the requirements of NEI 94-01, Revision 2-A, SER Section 3.1.1.2, GNPP 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.

Table 3.9.1-1 – NEI 94-01, Revision 2-A, Limitations and Conditions	
Limitation/Condition (From Section 4.0 of SE)	**GNPP Response**
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 the applicability of NEI 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable. GNPP was not licensed under 10 CFR 52.

3.10 Conclusion

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

NEI 94-01, Revision 2-A, dated October 2008, describes an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporates the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years. NEI 94-01, Revision 2-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. GNPP is adopting the guidance of NEI 94-01, Revision 2-A, for the GNPP 10 CFR 50, Appendix J testing program plan.

Based on the previous ILRTs conducted at GNPP, 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 GNPP inspection programs:

- ASME Section XI, IWE
- ASME Section XI, IWL
- Tendon Surveillance Program (TS 5.5.6)
- Maintenance Rule Structural Assessment and Monitoring Program
- Safety-Related Coatings Program

This experience is supplemented by risk analysis studies, including the GNPP risk analysis provided in Attachment 2 of this submittal. The risk assessment concludes that increasing the ILRT interval on a permanent basis to a one-in-fifteen-year frequency is not considered to be significant since it represents only a small change in the GNPP risk profile.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

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

10 CFR 50.36, Technical Specifications, provides the regulatory requirements for the content required in a plant's TS. 10 CFR 50.36(c)(5), "Administrative controls," requires that "provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner" will be included in the plant's TS.

10 CFR 50, Appendix J, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants," Option B, Section V.B, "Implementation" subparagraph 3, requires that the regulatory guide or implementation document used to develop a performance-based leakage-testing program be included by general reference in the plant TS. The Appendix J Testing Program is included in the Administrative Controls section of the GNPP TS as TS 5.5.15, Primary Containment Leakage Rate Testing Program. This LAR does not remove this administrative control requirement, but simply revises the administrative controls in TS 5.5.15 to include extending the frequency for performing the Type A ILRT to 15 years in accordance with NEI 94-01, Revision 2-A. Therefore, the 10 CFR 50.36 requirement continues to be met by this change.

The requirements to perform testing of the primary reactor containment are set forth in 10 CFR 50.54(o) and 10 CFR 50, Appendix J. Both of these CFR sections address criteria established in 10 CFR 50, Appendix A, General Design Criteria (GDC): GDC 50 (Containment Design Basis); GDC 51 (Fracture Prevention of Containment Pressure Boundary); GDC 52 (Capability for Containment Leakage Rate Testing); and, GDC 53 (Provisions for Containment Testing and Inspection). EGC has determined that the proposed change does not require any additional exemptions or relief from regulatory requirements and does not affect conformance with any GDC as described in the Updated Final Safety Analysis Report (UFSAR). However, this change does propose an extension of the frequency for performance of the Type A ILRT.

10 CFR 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of 10 CFR Part 50, Appendix J. Appendix J specifies containment leakage testing requirements, including the types required to ensure the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. In addition, Appendix J discusses leakage rate acceptance criteria, test methodology, frequency of testing and reporting requirements for each type of test.

The adoption of the Option B performance-based containment leakage rate testing for Type A, Type B and Type C testing did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, the test frequency is based upon an evaluation that reviewed "as-found" leakage history to determine the frequency for leakage testing, which provides assurance that leakage limits will be maintained. The change to the Type A test frequency did not directly result in an increase in containment leakage.

EPRI TR-1009325, Revision 2A (Reference 15), provided a risk impact assessment for optimized ILRT intervals up to 15 years, utilizing current industry performance data and risk informed guidance. NEI 94-01, Revision 2-A, Section 9.2.3.1 states that Type A ILRT intervals of up to 15 years are allowed by this guideline. The Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, EPRI Report 1018243 (formerly TR-1009325, Revision 2A) (Reference 15) indicates that, in general, the risk impact associated with ILRT interval extensions for intervals up to 15 years is small. However, plant-specific confirmatory analyses are required.

The NRC staff reviewed NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2A. For NEI TR 94-01, Revision 2-A, the NRC staff determined that it described an acceptable approach for implementing the optional performance-based requirements of 10 CFR 50, Appendix J, Option B. This guidance includes provisions for extending Type A ILRT intervals up to 15 years and incorporates the regulatory positions stated in RG 1.163 (Reference 3). The NRC staff finds that the Type A testing methodology as described in ANSI/ANS-56.8-2002 (Reference 2), and the modified testing frequencies recommended by NEI TR 94-01, Revision 2-A, serve to ensure continued leakage integrity of the containment structure. Type B and Type C testing ensures that individual penetrations are essentially leak tight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.

For EPRI Report No. 1009325, Revision 2A (Reference 15), a risk-informed methodology using plant-specific risk insights and industry ILRT performance data to revise ILRT surveillance frequencies, the NRC staff finds that the proposed methodology satisfies the key principles of risk-informed decision-making applied to changes to TS as delineated in RG 1.177, An Approach to Plant-Specific, Risk-Informed Decision making: Technical Specifications (Reference 23) and RG 1.174 (Reference 4). The NRC staff, therefore, found that this guidance was acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to the limitations and conditions noted in Section 4.2 of the SER.

Any applicant may reference NEI TR 94-01, Revision 2-A, dated October 2008, as modified by the associated SER and approved by the NRC, in a licensing action to satisfy the requirements of Option B to 10 CFR 50, Appendix J.

4.2 Precedent

This LAR is similar in nature to the following license amendments for extending the Type A test frequency to 15 years as previously authorized by the NRC:

- Nine Mile Point Nuclear Station, Unit 2 (Reference 24)
- Arkansas Nuclear One , Unit 2 (Reference 25)
- Palisades Nuclear Plant (Reference 26)
- Virgil C. Summer Nuclear Station, Unit 1 (Reference 27)
- Monticello Nuclear Generating Plant (MNGP) (Reference 28)

4.3 No Significant Hazards Consideration

Exelon Generation Corporation, LLC (EGC) proposes to amend the Technical Specifications (TS) 5.5.15, "Primary Containment Leakage Rate Testing Program," for R. E. Ginna Nuclear Power Plant (GNPP), to allow permanent extension of the Type A integrated leak rate test (ILRT) testing interval. The extension is based on the adoption of the Nuclear Energy Institute (NEI) 94-01, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 2-A.

Specifically, the proposed change revises GNPP TS 5.5.15 by replacing the reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," with a reference to NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 2-A

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

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

   Response: No.

   The proposed activity involves the revision of the R. E. Ginna Nuclear Power Plant (GNPP) Technical Specification (TS) 5.5.15, "Primary Containment Leakage Rate Testing Program," to allow the extension of the Type A Integrated Leakage Rate Test (ILRT) containment test interval to 15 years. Per the guidance provided in Nuclear Energy Institute (NEI) 94-01, Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J, Revision 2-A, the current Type A test interval of 10 years would be extended on a permanent basis to no longer than 15 years from the last Type A test.

   The proposed interval extensions do not involve either a physical change to the plant or a change in the manner in which 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.

   The change in 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, based on the probabilistic risk assessment (PRA) is 0.29 person-Roentgen equivalent man (rem)/year. Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2A states that a very small population dose is defined as an increase of less than 1.0 person-rem per year or less than 1 percent of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This is consistent with the Nuclear Regulatory Commission (NRC) Final

Safety Evaluation which endorsed NEI 94-01 and EPRI Report No. 1009325, Revision 2A. Moreover, the risk impact when compared to other severe accident risks is negligible. Therefore, the proposed extension does not involve a significant increase in the probability of an accident previously evaluated.

In addition, as documented in NUREG-1493, “Performance-Based Containment Leak-Test Program,” dated September 1995, Types B and C tests have identified a very large percentage of containment leakage paths, and the percentage of containment leakage paths that are detected only by Type A testing is very small. The GNPP Type A test history supports this conclusion.

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/or 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 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” Containment Maintenance Rule Inspections, Containment Coatings Program and TS 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 (ILRT). Based on the above, the proposed test interval extensions do not significantly increase the consequences of an accident previously evaluated.

This proposed amendment also deletes the exception previously granted to allow one-time extension of the ILRT test frequency for GNPP. Specifically, TS 5.5.15, item a. is deleted, as it requires the first Type A test performed after May 31, 1996, to be performed by May 31, 2011. This exception was included in the TS for one-time testing activities that would have already taken place by the time this amendment is approved; therefore, deletion is solely an administrative action that has no effect on any component and no impact on how the unit is operated.

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.

The proposed amendment to the GNPP TS 5.5.15 involves the extension of the GNPP Type A containment test interval from 10 years to 15 years. The containment and the testing requirements to periodically demonstrate the integrity of the containment exist to ensure the plant’s ability to mitigate the

consequences of an accident; thereby, do not involve any accident precursors or initiators.

The proposed change does not involve a physical modification to the plant (i.e., 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.

This proposed amendment also deletes the exception previously granted to allow one-time extension of the ILRT test frequency for GNPP. Specifically, TS 5.5.15, item a. is deleted, as it requires the first Type A test performed after May 31, 1996, to be performed by May 31, 2011. This exception was included in the TS for one-time testing activities that would have already taken place by the time this amendment is approved; therefore, deletion is solely an administrative action that has no effect on any component and no impact on how the unit is operated.

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 amendment to TS 5.5.15 involves the extension of the GNPP Type A containment test interval to 15 years. This amendment does not alter the manner in which safety limits, limiting safety system set points, or limiting conditions for operation are determined. The specific requirements and conditions of the TS Containment Leak 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 TS is maintained.

   The proposed change involves the extension of the interval between Type A containment leak rate tests for GNPP. The proposed surveillance interval extension is bounded by the 15-year ILRT interval currently authorized within NEI 94-01, Revision 2-A. Industry experience supports the conclusion that Types B and C testing detects a large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is small. The containment inspections performed in accordance with Option B to 10 CFR 50, Appendix J and the overlapping inspection activities performed as part of ASME Section XI, and the TS serve to provide a high degree of assurance that the containment would not degrade in a manner that is detectable only by Type A testing. The combination of these factors ensures that the margin of safety in the plant safety analysis is maintained. The design, operation, testing methods and acceptance criteria for Types A, B, and C containment leakage tests specified in applicable codes and standards would continue to be met, with the acceptance of this proposed change, since these are not affected by changes to the Type A test intervals.

In addition, this proposed amendment also deletes the exception previously granted to allow one-time extension of the ILRT test frequency for GNPP. Specifically, TS 5.5.15, item a. is deleted, as it requires the first Type A test performed after May 31, 1996, to be performed by May 31, 2011. This exception was included in the TS for one-time testing activities that would have already taken place by the time this amendment is approved; therefore, deletion is solely an administrative action that has no effect on any component and no impact on how the unit is operated

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

Based on the above, EGC 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.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 REFERENCES

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

2. ANSI/ANS 56.8-2002, “Containment System Leakage Testing Requirements,” LaGrange Park, Illinois, November 2002

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

4. 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,” January 2018

5. Regulatory Guide 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” March 2009

6. NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 21, 1995

7. NUREG-1493, “Performance-Based Containment Leak-Test Program,” January 1995

8. EPRI TR-104285, “Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals,” August 1994

9. NEI 94-01, Revision 3-A, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 2012

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

11. Letter from S. Bahadur (NRC) to B. Bradley (NEI), "Final Safety Evaluation of Nuclear Energy Institute (NEI) Report 94-01, Revision 3, Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J (TAC No. ME2164)," dated June 8, 2012

12. Letter from A. R. Johnson (NRC) to Dr. R C. Mecredy (GNPP), “Issuance of Amendment No. 61 to Facility Operating License No. DPR-18 Regarding Implementation of the Amended Regulation 10 CFR Part 50, Appendix J, Option B, to Provide a Performance Based Option for Leakage-Rate Testing of Containment–R.E. Ginna Nuclear Power Plant (TAC Nos. M89516, M89559, M92320, M92963, …, M92969, M93579, M93071, M93708, and M93928).” Dated February 13, 1996

13. ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements," dated August 4, 1994

14. Letter from P. D. Milano (NRC) to Mary G. Korsnick (GNPP), "Issuance of Amendment No. 93 to Renewed Facility Operating License No. DPR-18 Regarding One-Time Extension of Containment Integrated Leakage Rate Test Interval – R. E. Ginna Nuclear Power Plant (TAC No. MC6375)," dated December 8, 2005

15. Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325. EPRI, Palo Alto, CA: October 2008, 1018243

16. 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, November 2001

17. Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant) to NRC (Document Control Desk), Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, (Docket No. 50-317), dated March 27, 2002

18. Ginna Nuclear Station, G1-LAR-002, "Responses to Request for Additional Information Regarding LAR for Adopting TSTF-425," February 2016

19. Regulatory Letter from Mr. Joseph E. Pacher (Ginna) to NRC (Document Control Desk), "License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," dated March 28, 2013 (ML13093A064)

20. NUREG-2122, "Glossary of Risk-Related Terms in Support of Risk-Informed Decision making," dated November 2013 (ML13311A353)

21. Letter from NRC (R. A. Auluck) to GNPP (Dr. R. C. Mecredy), "Issuance of Renewed Facility Operating License No. DPR-18 for R. E. Ginna Nuclear Power Plant," dated May 19, 2004 (ML 041400502)

22. NUREG-1801, Generic Aging Lessons Learned (GALL Report)

23. Regulatory Guide 1.177, Revision 1, An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications, May 2011

24. Letter from R. Guzman (NRC) to S. Belcher (NMP), "Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment RE: Extension of Primary Containment Integrated Leakage Rate Testing Interval (TAC No. ME1650)," dated March 30, 2010

25. Letter from N Kalyanam (NRC) to Vice President, Operations (Entergy), "Arkansas Nuclear One, Unit 2 – Issuance of Amendment RE: Technical Specifications Change to Extend Type A Test Frequency to 15 Years (TAC No. ME4090)", dated April 7, 2011

26. Letter from M. Chawala (NRC) to Vice President, Operations (Entergy), "Palisades Nuclear Plant – Issuance of Amendment to Extend the Containment Type A Leak Rate Test Frequency to 15 Years (TAC No. ME5997)", dated April 23, 2012

27. Letter from S. Williams (NRC) to T.D. Gatlin (VCNS), “Virgil C. Summer Nuclear Station, Unit 1 – Issuance of Amendment Extending Integrated Leak Rate Test Interval (TAC No. MF1385), dated February 5, 2014

28. Letter from R. F. Kuntz (NRC) to P.A. Gardner (MNGP), “Monticello Nuclear Generating Plant – Issuance of Amendment Re: Technical Specification 5.5.11, “Primary Containment Leakage Rate Testing Program (CAC No. MF7359)”, dated April 25, 2017 (ML17103A235)

29. RG 1.174, Revision 2, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” May 2011

30. Letter from J. Pacher (Constellation Energy) to NRC (Document Control Desk), “First Interval IWE/IWL Containment Program Submittal of Relief Request Number 22,” dated November 21, 2008

ATTACHMENT 2

Markup of Technical Specifications Pages

R. E. Ginna Nuclear Power Plant

Renewed Facility Operating License No. DPR-18

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<u>Revised Technical Specifications Pages</u>

<u>TS Pages</u>

5.5-11

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

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

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

5.5.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to~~ NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance- Based Option of 10 CFR 50, Appendix J~~":~~ ,”Revision 2-A, dated October 2008.

~~a. Section 9.2.3: The first Type A test performed after the May 31, 1996 Type A test shall be performed by May 31, 2011.~~

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 60 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq P_a$

ATTACHMENT 3

Retyped Version of Technical Specifications Pages

R. E. Ginna Nuclear Power Plant

Renewed Facility Operating License No. DPR-18

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

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

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

5.5.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance- Based Option of 10 CFR 50, Appendix J," Revision 2-A, dated October 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 60 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq P_a$.

ATTACHMENT 4

Ginna Evaluation of Risk Significance of Permanent ILRT Extension

R. E. Ginna Nuclear Power Plant

Renewed Facility Operating License No. DPR-18

Docket No. 50-244

PRA APPLICATION NOTEBOOK

G1-LAR-004

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

Revision 2

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Document Review			
RM DOCUMENTATION NO.	G1-LAR-004	**REV:** 2	**PAGE NO.** 2

STATION: Ginna
UNIT(S) AFFECTED: UNIT 1

TITLE: Evaluation of Risk Significance of Permanent ILRT Extension

SUMMARY (Include UREs incorporated):

GGNGS is pursuing a License Amendment Request (LAR) to permanently extend the Type A Integrated Leak Rate Test (ILRT) to 15 years.

The purpose of this document is to provide an assessment of the risk associated with implementing a permanent extension of the GGNGS Unit 1 and Unit 2 containment ILRT and DWBT interval to 15 years.

URE(s) Impacted: None

Number of pages: Total 54 pages, including this page.

RM Document Level: Category 1, per ER-AA-600-1012

[X] Review Required after periodic Update

[X] Internal RM Documentation [ ] External RM Documentation

Electronic Calculation Data Files: N/A

Method of Review: [X] Detailed [ ] Alternate [ ] Review of External Document

This RM documentation supersedes: NA in its entirety

	Name		Signature		Date
Prepared by:	Justin Sattler	/		/	
Reviewed by:	Craig Matos	/		/	
Approved by:	Gene Kelly	/		/	1/31/2019

Do any ASSUMPTIONS / ENGINEERING JUDGEMENTS require later verification? **[ ] Yes [X] No** Tracked By: AT#, URE# etc.):

REVISION RECORD SUMMARY

Revision	Revision Summary
0	Initial Issue
1	Editorial comments incorporated
2	Editorial comments incorporated

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) from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages for the Ginna Nuclear Power Plant (GNPP). The risk assessment follows the guidelines from NEI 94-01, Revision 3-A [Reference 1], 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 (15 years in this extension analysis). 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), 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 GNPP.

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 changes for a 15-year interval.

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. Since the Type A test does not impact CDF, 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 small population dose is defined as an increase from the baseline interval (3 tests per 10 years to 1 test in 15 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, November 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: Surry Unit 1*, Main Report NUREG/CR-4551, SAND86-1309, Volume 3, Revision 1, Part 1, October 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. Ginna Nuclear Power Plant Probabilistic Risk Assessment, “Quantification Notebook,” G1-PRA-014, Revision 1, March 2016.

18. R. E. Ginna NPP, “Fire PRA Notebook Plant Response Model (PRM),” G1-PRM-F001, Revision 4, March 2017.
19. Application for Renewed Operating License, Appendix E – Environmental Report, R.E. Ginna Nuclear Power Plant.
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. NUREG/CR-2728, Interim Reliability Evaluation Program Procedures Guide, March 3, 1983; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852.
28. Letter L-14-121, ML14111A291, FENOC Evaluation of the Proposed Amendment, Beaver Valley Power Station, Unit Nos. 1 and 2, April 2014.
29. Technical Letter Report ML112070867, Containment Liner Corrosion Operating Experience Summary, Revision 1, August 2011.
30. Ginna Nuclear Power Plant Probabilistic Risk Assessment, “PRA Large Early Release Analysis Notebook,” G1-PRA-015, Revision 2, March 2016.
31. Letter from R. S. Mecredy to G. S. Vissing, “Resolution of Generic Letter 87-02, Supplement 1 and Generic Letter 88-20, Supplements 4 and 5 (Seismic Events Only),” January 31, 1997.
32. Ginna NFPA 805 LAR, Attachment W, Revision 2.
33. R. E. Ginna Nuclear Power Plant Technical Procedure STP-O-R-6.0, “Containment Integrated Leakage Rate Test.”
34. WORKCOMP-20110628-00018, Performance of Technical Procedure STP-O-R-6.0, “Containment Integrated Leakage Rate Test.”
35. 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.”

36. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.
37. ASME/ANS RA-Sa-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," 2009.
38. NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, November 2008.
39. Ginna Nuclear Power Plant, "Updated Final Safety Analysis Report," Revision 27, November 2017.
40. NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Revision 1, June 2010.
41. NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," August 2012.
42. Letter from D. Render to B. Hanson, ML15271A101, R.E. Ginna Nuclear Power Plant, "R.E. Ginna Nuclear Power Plant – Issuance of Amendment Regarding Transition to Risk Informed, Performance-Based Fire Protection Program in Accordance with Title 10 of the Code of Federal Regulations Section 50.48(c) (CAC NO. MF1393)," November 23, 2015.
43. 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.
44. R.E. Ginna Nuclear Power Plant, IPEEE Seismic Evaluation Report, January 1997.
45. Letter from A. Dimitriadis to B. Hanson, R.E. Ginna Nuclear Power Plant, LLC – Temporary Instruction 2515/191 Inspection Report 05000244/2016011, December 2016, ML16337A092.
46. R.E. Ginna Nuclear Power Plant, Individual Plant Examination of External Events (IPEEE), 180-Day Response to Generic Letter 88-20, Supplement 4, December 1991, ML17262A711.
47. R.E. Ginna Nuclear Power Plant, Individual Plant Examination of External Events (IPEEE), High Winds, External Floods and Transportation Accidents, December 1998, Letter RG016285.
48. Letter from M. Halter to B. Hanson, R.E. Ginna Nuclear Power Plant – Safety Evaluation Regarding Implementation of Mitigating Strategies and Reliable Spent Fuel Pool Instrumentation Related to Orders EA-12-049 and EA-12-051 (CAC NOs. MF1152 and MF1147), July 2016, ML16124A038.
49. "Calvert Cliffs Nuclear Power Plant, Units 1 and 2 - License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b'," February 25, 2016, ML16060A223.
50. "U.S. Nuclear Regulatory Commission, Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 12, 2012, ML12053A340.
51. "Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-

Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 31, 2014, ML14099A196.

52. "R.E. Ginna Nuclear Power Plant - Staff Assessment of Information Provided Pursuant to Title 10 of the *Code of Federal Regulations* Part 50, Section 50.54(f), Seismic Hazard Reevaluations for Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident (TAC No. MF3972)," June 11, 2015, ML15153A026.
53. "High Frequency Supplement to Seismic Hazard Screening Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," December 4, 2015, ML15338A003.
54. "Staff Review of High Frequency Confirmation Associated with Reevaluated Seismic Hazard in Response to March 12, 2012 50.54(f) Request for Information," February 18, 2016, ML15364A544.
55. EPRI 3002000709, Final Report, Seismic Probabilistic Risk Assessment Implementation Guide, Electric Power Research Institute, December 2013.
56. "Risk Assessment of Operational Events, Volume 2 – External Events – Internal Fires – Internal Flooding – Seismic – Other External Events – Frequencies of Seismically-Induced LOOP Events (RASP Handbook)", Revision 1.02, US Nuclear Regulatory Commission, November 2017.
57. Letter from Mr. Joseph E. Pacher (Ginna) to Document Control Desk (NRC), dated March 28, 2013, "License Amendment Request Pursuant to 10 CFR 50.90: Adoption of NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," ML13093A064.
58. Ginna Nuclear Station, G1-LAR-002, "Responses to Request for Additional Information Regarding LAR for Adopting TSTF-425," February 2016.
59. Report 032299-RPT-002, "Risk Management Finding Level F&O Technical Review," Revision 0, August 2017.
60. Exelon Procedure ER-AA-600-1068, "Fire PRA Risk Evaluation Support for NFPA 805 Program," Revision 3.

4.0 ASSUMPTIONS AND LIMITATIONS

The following assumptions were used in the calculation:

- The technical adequacy of the GNPP PRA [Reference 17] is either consistent with the requirements of Regulatory Guide 1.200, or where gaps exist, the gaps have been addressed, as detailed in Attachment 1.
- The GNPP Level 1 and Level 2 internal events PRA models provide representative results.
- It is appropriate to use the GNPP internal events PRA model to effectively describe the risk change attributable to the ILRT extension. A study is done in Section 5.2.7 to show the effect of including external event models for the ILRT extension. The additional risk from a Seismic PRA [Reference 35] and the Fire PRA [Reference 18] are used for this analysis.
- Accident classes describing radionuclide release end states are defined consistent with EPRI methodology [Reference 24].
- 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 [Reference 24].
- While precise numbers are maintained throughout the calculations, some values have been rounded when presented in this report. Therefore, rounding differences may result in table summations.

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 GNPP. 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.

<u>NUREG-1273 [Reference 12]</u>

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.

<u>NUREG/CR-4330 [Reference 13]</u>

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.”

<u>EPRI TR-105189 [Reference 14]</u>

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.

<u>NUREG-1493 [Reference 6]</u>

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.

<u>EPRI TR-104285 [Reference 2]</u>

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 0.02 person-rem per year…"

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

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 Surry. The ex-plant calculation can be delineated to total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551. With the GNPP Level 2 model end-states assigned to one of the NUREG/CR-4551 APBs, it is considered adequate to represent GNPP. (The meteorology and site differences other than population are assumed not to play a significant role in this evaluation.)

<u>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]</u>

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.

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

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.

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

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 GNPP 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 GNPP ILRT Extension Risk Assessment includes the following:

- CDF Model results [Reference 17]
- LERF Model results [Reference 30]
- Release category definitions used in the Level 2 Model [Reference 19]
- Dose within a 50-mile radius [Reference 19]
- ILRT results to demonstrate adequacy of the administrative and hardware issues [Reference 34]

GNPP Model

The Internal Events PRA Model that is used for GNPP is characteristic of the as-built plant. The current Level 1, LERF, and Level 2 model (model name GN016A) is a linked fault tree model [Reference 17]. The total CDF is 1.08E-5/year; the total LERF is 4.34E-7 [Reference 17]. Table 5-1 and Table 5-2 provide a summary of the Internal Events CDF and LERF results for GNPP PRA Model.

Fire CDF is 2.92E-5/year, and Fire LERF is 6.31E-7/year [Reference 18]. The Seismic PRA results from Generic Issue 199 yields a CDF of 6.55E-6/year [Reference 36]. Refer to Section 5.2.7 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	5.58E-06
Transients	3.72E-06
LOCAs	5.87E-07
SGTR	4.00E-07
RPV Rupture	2.90E-08
ISLOCA	9.40E-09
Loss of Offsite Power	4.83E-07
Total Internal Events CDF	**1.08E-05**

Table 5-2 – Internal Events LERF

Internal Events	Frequency (per year)
Internal Floods	7.57E-09
Transients	7.99E-09
LOCAs	8.51E-10
SGTR	4.12E-07
RPV Rupture	3.91E-11
ISLOCA	5.63E-09
Loss of Offsite Power	3.04E-10
Total Internal Events LERF	**4.34E-07**

Population Dose Calculations

The population dose calculation was reported in the License Renewal Application [Reference 19]. Table 5-3 presents dose exposures calculated from methodology described in Reference 1 and data from Reference 19. Reference 19 "Intact Containment" Release Category corresponds to EPRI Accident Class 1. "LOCI" (Loss of Containment Isolation) Release Category corresponds to EPRI Accident Class 2. Since they are not associated with other classes, four containment end-states correspond to EPRI Accident Class 7 ("Late Failure Global," "Late Failure Small," "HPRCS," and "LPRCS" Containment Failure Types); the EPRI Accident Class 7 dose is calculated via a weighted average using the frequencies provided in Reference 19. The "TISGTR" Release Category (this SGTR release category is used instead of other SGTR release categories because it represents unscrubbed SGTR dose and every LERF SGTR cutset includes failure of scrubbing) and "ISLOCA" Release Category correspond to EPRI Accident Class 8; dose used in this analysis is weighted via the ISLOCA and SGTR frequencies in this calculation. Class 3a and 3b population dose values are calculated from the Class 1 population dose and represented as $10L_a$ and $100L_a$, respectively, as guidance in Reference 1 dictates.

Table 5-3 – Population Dose

Accident Class	Description	Release (person-rem)
1	Containment Remains Intact	2.27E+04
2	Containment Isolation Failures	3.38E+06
3a	Independent or Random Isolation Failures SMALL	2.27E+05[1]
3b	Independent or Random Isolation Failures LARGE	2.27E+06[2]
4	Isolation Failure in which pre-existing leakage is not dependent on sequence progression. Type B test Failures	n/a
5	Isolation Failure in which pre-existing leakage is not dependent on sequence progression. Type C test Failures	n/a
6	Isolation Failure that can be verified by IST/IS or surveillance	n/a
7	Containment Failure induced by severe accident	9.61E+05
8	Accidents in which containment is by-passed	4.89E+06

1. $10 * L_a$
2. $100 * L_a$

Release Category Definitions

Table 5-4 defines the accident classes used in the ILRT extension evaluation, which is consistent with the EPRI methodology [Reference 24]. 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 5-4 – EPRI Containment Failure Classification [Reference 24]

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 5-4, 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 delta LERF 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 GNPP, as detailed in Section 5.2, involves subtracting LERF risk from the CDF that is applied to Class 3b because this portion of LERF is unaffected by containment integrity. 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 1009325 [Reference 24] 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 5-5.

The analysis performed examined GNPP-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 1009325, Class 1 sequences [Reference 24]).
- 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 1009325, Class 3 sequences [Reference 24]).
- Accident sequences involving containment bypassed (EPRI 1009325, Class 8 sequences [Reference 24]), large containment isolation failures (EPRI 1009325, Class 2 sequences [Reference 24]), and small containment isolation “failure-to-seal” events (EPRI 1009325, Class 4 and 5 sequences [Reference 24]) 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 5-5 – 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 (Early and Late)
8	Bypass (Interfacing System LOCA)
CDF	All CET End States (Including Very Low and No Release)

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

Step 1 - Quantify the baseline risk in terms of frequency per reactor year for each of the accident classes presented in Table 5-5.

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 1009325 [Reference 24]). 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 5-5 were developed for GNPP by first determining the frequencies for Classes 1, 2, 6, 7, and 8. Table 5-6 presents the grouping of each release category in EPRI Classes based on the associated description. Table 5-7 presents the frequency and EPRI category for each sequence and the totals of each EPRI classification. Table 5-8 provides a summary of the accident sequence frequencies that can lead to radionuclide release to the public and have been derived consistent with the NEI Interim Guidance [Reference 3] and the definitions of accident classes 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 section. Therefore, minor differences may occur if the calculations in these sections are followed explicitly.

The total CDF is 1.08E-5 and LERF is 4.34E-7 [Reference 17].

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

$$Freq_{class3a} = P_{class3a} * (CDF - LERF)$$
$$= \frac{2}{217} * (1.08\text{E-}5 - 4.34\text{E-}7) = 9.56\text{E-}8$$

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:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF)$$
$$= \frac{.5}{218} * (1.08\text{E-}5 - 4.34\text{E-}7) = 2.38\text{E-}8$$

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). The intact frequency accounts for 62.2% of CDF [Reference 30], which means Class 1 frequency is 6.72E-6. The frequency per year is initially determined from the EPRI Accident Class 1 frequency listed in Table 5-7 and then subtracting the EPRI Class 3a and 3b frequency (to preserve total CDF), calculated below:

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

Class 2 Sequences. This group consists of core damage accident progression bins with large containment isolation failures. This is determined from flag CTAZISO_LERF, the contribution of large containment isolation failure flag for LERF. Since this event is in cutsets that contribute 0.612% of LERF, the Class 2 contribution is 2.66E-9. The frequency per year for these sequences is obtained from the EPRI Accident Class 2 frequency listed in Table 5-7.

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 frequency listed in Table 5-7.

Class 8 Sequences. This group consists of all core damage accident progression bins in which containment is bypassed via ISLOCA or SGTR. Since the ISLOCA initiator is in cutsets that contribute 1.30% of LERF, its Class 8 contribution is 5.63E-9. Since the SGTR initiators are in cutsets that contribute 94.8% of LERF, its Class 8 contribution is 4.12E-7. For this analysis, the frequency is determined from the EPRI Accident Class 8 frequency listed in Table 5-7.

LERF quantification is distributed into EPRI categories based on release categories. Table 5-6 shows this distribution.

Table 5-6 – Release Category Frequencies

Containment End State	EPRI Category	Frequency (/yr)
Intact Containment	1	6.72E-06
Large Isolation Failure	2	2.66E-09
Failures Induced by Phenomena	7	3.66E-06
ISLOCA	8	5.63E-09
SGTR	8	4.12E-07

Table 5-7 – Accident Class Frequencies

EPRI Category	Frequency (/yr)
Class 1	6.72E-06
Class 2	2.66E-09
Class 6	ε[1]
Class 7	3.66E-06
Class 8	4.17E-07
Total (CDF)	1.08E-05

1. ε represents a probabilistically insignificant value.

Table 5-8 – Baseline Risk Profile

Class	Description	Frequency (/yr)
1	No containment failure	6.60E-06[2]
2	Large containment isolation failures	2.66E-09
3a	Small isolation failures (liner breach)	9.56E-08
3b	Large isolation failures (liner breach)	2.38E-08
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 (early and late)	3.66E-06
8	Containment bypass	4.17E-07
	Total	1.08E-05

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)

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. Table 5-3 provides population dose for each EPRI accident class. Table 5-9 provides a correlation of GNPP population dose to 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:

$$EPRI\ Class\ 3a\ Population\ Dose = 10 * 2.27E{+}4 = 2.27E{+}5$$

$$EPRI\ Class\ 3b\ Population\ Dose = 100 * 2.27E{+}4 = 2.27E{+}6$$

Table 5-9 – Mapping of Population Dose to EPRI Accident Class

EPRI Category	Frequency (/yr)	Dose (person-rem)
Class 1	6.60E-06	2.27E+04
Class 2	2.66E-09	3.38E+06
Class 7	3.66E-06	9.61E+05
Class 8	4.17E-07	4.89E+06

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).

<u>Risk Impact Due to 10-Year Test Interval</u>

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 and population dose rates are calculated as follows:

$$Freq_{Class3a10yr} = \frac{10}{3} * \frac{2}{217} * (CDF - LERF) = \frac{10}{3} * \frac{2}{217} * 1.04\text{E-}5 = 3.19\text{E-}7$$

$$Freq_{Class3b10yr} = \frac{10}{3} * \frac{.5}{218} * (CDF - LERF) = \frac{10}{3} * \frac{.5}{218} * 1.04\text{E-}5 = 7.93\text{E-}8$$

$$PopDoseRate_{Class3a10yr} = Freq_{Class3a10yr} * PopDose_{Class3a} = 3.19\text{E-}7 * 2.27\text{E+}5 = 7.23\text{E-}2$$

$$PopDoseRate_{Class3b10yr} = Freq_{Class3b10yr} * PopDose_{Class3b} = 7.93\text{E-}8 * 2.27\text{E+}6 = 1.80\text{E-}1$$

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

Table 5-10 – 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]	6.32E-06	58.52%	2.27E+04	1.44E-01
2	Large containment isolation failures	2.66E-09	0.02%	3.38E+06	8.98E-03
3a	Small isolation failures (liner breach)	3.19E-07	2.95%	2.27E+05	7.23E-02
3b	Large isolation failures (liner breach)	7.93E-08	0.73%	2.27E+06	1.80E-01
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 (early and late)	3.66E-06	33.91%	9.61E+05	3.52E+00
8	Containment bypass	4.17E-07	3.86%	4.89E+06	2.04E+00
	Total	1.08E-05			5.97E+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 and population dose rates are calculated as follows:

$$Freq_{Class3a15yr} = \frac{15}{3} * \frac{2}{217} * (CDF - LERF) = 5 * \frac{2}{217} * 1.04\text{E-}5 = 4.78\text{E-}7$$

$$Freq_{Class3b15yr} = \frac{15}{3} * \frac{.5}{218} * (CDF - LERF) = 5 * \frac{.5}{218} * 1.04\text{E-}5 = 1.19\text{E-}7$$

$$PopDoseRate_{Class3a15yr} = Freq_{Class3a15yr} * PopDose_{Class3a} = 4.78\text{E-}7 * 2.27\text{E+}5 = 1.08\text{E-}1$$

$$PopDoseRate_{Class3b15yr} = Freq_{Class3b15yr} * PopDose_{Class3b} = 1.19\text{E-}7 * 2.27\text{E+}6 = 2.70\text{E-}1$$

The results of the calculation for a 15-year interval are presented in Table 5-11.

Table 5-11 – 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]	6.12E-06	56.68%	2.27E+04	1.39E-01
2	Large containment isolation failures	2.66E-09	0.02%	3.38E+06	8.98E-03
3a	Small isolation failures (liner breach)	4.78E-07	4.42%	2.27E+05	1.08E-01
3b	Large isolation failures (liner breach)	1.19E-07	1.10%	2.27E+06	2.70E-01
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 (early and late)	3.66E-06	33.91%	9.61E+05	3.52E+00
8	Containment bypass	4.17E-07	3.86%	4.89E+06	2.04E+00
	Total	1.08E-05			6.09E+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

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 and no equipment in the intermediate building is credited in the CDF model at GNPP, the ILRT extension does not impact CDF [Reference 37]. Therefore, the relevant risk-impact metric is LERF.

For GNPP, 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 10-year test interval from Table 5-10, the Class 3b frequency is 7.93E-8/year; based on a 15-year test interval from Table 5-11, the Class 3b frequency is 1.19E-7/year. Thus, the increase in the overall probability of LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is 9.52E-8/year. Similarly, the increase due to increasing the interval from 10 to 15 years is 3.96E-8/year. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF meets the 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 5-12 summarizes these results.

Table 5-12 – 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)	2.38E-08	7.93E-08	1.19E-07
ΔLERF (3 year baseline)		5.55E-08	9.52E-08
ΔLERF (10 year baseline)			3.96E-08

The increase in the overall probability of LERF due to Class 3b sequences is less than 10^{-7}. Therefore, the ΔLERF is considered very small [Reference 4].

NEI 94-01 [Reference 1] states that a 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 5-13, the results of this calculation meet the dose rate criteria.

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

ILRT Inspection Interval	10 Years	15 Years
ΔDose Rate (3 year baseline)	1.703E-01	2.920E-01
ΔDose Rate (10 year baseline)		1.217E-01
%ΔDose Rate (3 year baseline)	2.937%	5.034%
%ΔDose Rate (10 year baseline)		2.038%

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 5-14 shows the steps and results of this calculation.

Table 5-14 – Impact on CCFP due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	6.70E-06	6.64E-06	6.60E-06
f(ncf)/CDF	0.620	0.615	0.611
CCFP	0.380	0.385	0.389
ΔCCFP (3 year baseline)		0.514%	0.881%
ΔCCFP (10 year baseline)			0.367%

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 0.881%. Therefore, this increase is judged to be small.

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

An estimate of the likelihood and risk implications 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>

- Consistent with the Calvert Cliffs analysis, a half failure is assumed for basemat concealed liner corrosion due to the lack of identified failures (See Table 5-15, Step 1).
- In the 5.5 years following September 1996 when 10 CFR 50.55a started requiring visual inspection, there were three events where a through wall hole in the containment liner was identified. These are Brunswick 2 on 4/27/99, North Anna 2 on 9/23/99, and D. C. Cook 2 in November 1999. The corrosion associated with the Brunswick event is believed to have started from the coated side of the containment liner. Although GNPP has a different containment type, this event could potentially occur at GNPP (i.e., corrosion starting on the coated side of containment). Construction material embedded in the concrete may have contributed to the corrosion. The corrosion at North Anna is believed to have started on the uninspectable side of containment due to wood imbedded in the concrete during construction. The D. C. Cook event is associated with an inadequate repair of a hole drilled through the liner during construction. Since the hole was created during construction and not caused by corrosion, this event does not apply to this analysis. Based on the above data, there are corrosion events from the 5.5 years that apply to GNPP.
- 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 5-4, 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 5-15, 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 GNPP, the ILRT maximum pressure is 60 psig [References 33]. 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 5.3.1 (See Table 5-15, 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 5-15, Step 4).
- In the Calvert Cliffs analysis, it is noted that approximately 85% of the interior wall surface is accessible for visual inspections. 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 5-15, 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 5-15 – Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome (85%)	Containment Basemat (15%)
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: 0 Assume a half failure 0.5 / (70 x 5.5) = 1.30E-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 / 5.13E-04 average 5-10 / 1.30E-03 15 / 3.57E-03 15 year average = 1.61E-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.	0.71% (1 to 3 years) 4.14% (1 to 10 years) 9.66% (1 to 15 years)	0.18% (1 to 3 years) 1.04% (1 to 10 years) 2.42% (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

Table 5-15 – Steel Liner Corrosion Base Case

Step	Description	Containment Cylinder and Dome (85%)	Containment Basemat (15%)
6	Likelihood of non-detected containment leakage (Steps 3 x 4 x 5)	0.00071% (3 years) 0.71% x 1% x 10% 0.00414% (10 years) 4.18% x 1% x 10% 0.00966% (15 years) 9.66% x 1% x 10%	0.00018% (3 years) 0.18% x 0.1% x 100% 0.00104% (10 years) 1.04% x 0.1% x 100% 0.00242% (15 years) 2.42% 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 GNPP.

Table 5-16 – Total Likelihood on Non-Detected Containment Leakage Due to Corrosion for GNPP

Description
At 3 years: 0.00071% + 0.00018% = 0.00089%
At 10 years: 0.0041% + 0.00104% = 0.00517%
At 15 years: 0.00966% + 0.00242% = 0.01207%

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. 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 [Reference 29]. 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. For risk evaluation purposes, these five 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., 5/(66*17.1) = 4.43E-03) are bounded by the estimated historical flaw probability based on the two events in the 5.5 year period of the Calvert Cliffs analysis (i.e., 2/(70*5.5) = 5.19E-03) incorporated in the EPRI guidance [Reference 28].

5.2.7 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.

Ginna has transitioned to NFPA 805 licensing basis for fire protection and submitted a License Amendment Request (LAR) [Reference 42]. This transition includes performing a Fire PRA and installing modifications to reduce the fire-induced CDF and LERF to those reported in the NFPA 805 LAR.

The Fire PRA model was used to obtain the fire CDF and LERF values [Reference 18]. 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, LERF contributions from CDF are removed. The following shows the calculation for Class 3b:

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (2.92E\text{-}5 - 6.31E\text{-}7) = 6.55E\text{-}8$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (2.92E\text{-}5 - 6.31E\text{-}7) = 2.18\text{E-}7$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (2.92E\text{-}5 - 6.31E\text{-}7) = 3.28\text{E-}7$$

The IPEEE Seismic Evaluation Report does not result in an estimate of CDF [Reference 44]. Ginna submitted a seismic hazard screening report [Reference 50] to the NRC in accordance with the requirements of Near-Term Task Force (NTTF) Recommendation 2.1 Seismic [Reference 51]. Reference 50 confirmed the current GNPP seismic hazard (i.e., Ground Motion response Spectra (GMRS)) is bounded by the GNPP seismic capability (i.e., safe shutdown earthquake (SSE)) in the frequency range of one to ten Hz, except for a narrow band exceedance between 9 and 10 Hz. As a result, the NRC issued a staff assessment report [Reference 52] and concluded a seismic risk evaluation (i.e., seismic PRA or seismic margins assessment) is not required for GNPP. In Reference 52, the NRC also concluded that for NTTF Recommendation 2.1 Seismic, a spent fuel pool assessment was not required but a high-frequency confirmation was required. The licensee submitted a high frequency confirmation [Reference 53] that was accepted by the NRC in a separate staff assessment [Reference 54].

Since these updated evaluations do not include a quantitative evaluation of seismic risk (and were not required to), another analysis is used to estimate CDF. To estimate Seismic CDF (SCDF), the plant level HCLPF is used to convolve the corresponding failure probabilities as a function of seismic hazard level with the seismic hazard curve. For example, such a calculation approach was used by the NRC in their risk assessment of GI-199 [Reference 35], and Exelon used this method for a TSTF-505 submittal for Calvert Cliffs [Reference 49]. Further details of the SCDF calculation including the seismic hazard input (obtained from Reference 51 and shown in Table B-1) and seismic hazard intervals used in the analysis with their representative PGA and occurrence frequency (shown in Table B-2). The SCDF is estimated to be 3.86E-06.

Applying the internal event LERF/CDF ratio to the SCDF yields an estimated seismic LERF of 1.56E-7, as shown by the equation below.

$$\mathrm{LERF_{Seismic}} \approx \mathrm{CDF_{Seismic}} * \mathrm{LERF_{IE}} / \mathrm{CDF_{IE}} = 3.88\text{E-}6 * 4.34\text{E-}7 / 1.08\text{E-}5 = 1.56\text{E-}7$$

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

$$Freq_{class3b} = P_{class3b} * (CDF - LERF) = \frac{0.5}{218} * (3.88E\text{-}6 - 1.56E\text{-}7) = 8.53\text{E-}9$$

$$Freq_{class3b10yr} = \frac{10}{3} * P_{class3b} * (CDF - LERF) = \frac{10}{3} * \frac{0.5}{218} * (3.88E\text{-}6 - 1.56E\text{-}7) = 2.84\text{E-}8$$

$$Freq_{class3b15yr} = \frac{15}{3} * P_{class3b} * (CDF - LERF) = 5 * \frac{0.5}{218} * (3.88E\text{-}6 - 1.56E\text{-}7) = 4.27\text{E-}8$$

The fire and seismic 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 5-17.

Table 5-17 – GNPP 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	7.41E-08	2.47E-07	3.70E-07	2.96E-07
Internal Events	2.38E-08	7.93E-08	1.19E-07	9.52E-08
Combined	9.78E-08	3.26E-07	4.89E-07	3.91E-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 increase due to increasing the interval from 10 to 15 years is 1.63E-7; the total change in LERF due to increasing the ILRT interval from 3 to 15 years is 3.91E-7, which meets the guidance for small change in risk, as it exceeds 1.0E-7/yr and remains less than a 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 values are calculated below:

$$\mathrm{LERF} = \mathrm{LERF}_{\mathrm{internal}} + \mathrm{LERF}_{\mathrm{fire}} + \mathrm{LERF}_{\mathrm{seismic}} + \mathrm{LERF}_{\mathrm{class3Bincrease}}$$

$$\mathrm{LERF}_{10\mathrm{yr}} = 4.34\text{E-}7/\mathrm{yr} + 6.31\text{E-}7/\mathrm{yr} + 1.56\text{E-}7/\mathrm{yr} + 2.28\text{E-}7/\mathrm{yr} = 1.45\text{E-}6/\mathrm{yr}$$

$$\mathrm{LERF}_{15\mathrm{yr}} = 4.34\text{E-}7/\mathrm{yr} + 6.31\text{E-}7/\mathrm{yr} + 1.56\text{E-}7/\mathrm{yr} + 3.91\text{E-}7/\mathrm{yr} = 1.61\text{E-}6/\mathrm{yr}$$

Several conservative assumptions were made in this ILRT analysis, as discussed in Sections 4.0, 5.1.3, 5.2.1, and 5.2.4; therefore, the total change in LERF is considered conservative for this application. 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.

5.2.7.1 Screened External Hazards

Several “other” external events were evaluated in the GNPP IPEEE. Reference 46 evaluated hazards from high winds, external floods, and transportation accident. It concluded Ginna could withstand a 10^{-5} tornado within design limits, and a 10^{-6} tornado without structural failure. This evaluation meets line (4) of figure 1 of Generic Letter 88-20, Supplement 4 [Reference 46]. Since the time the IPEEE was performed, FLEX has been installed at GNPP to provide additional accident mitigation capabilities. Following inspection of Ginna’s mitigation strategies for beyond-design basis external events, NRC inspectors verified Exelon satisfactorily implemented appropriate elements of the FLEX strategy as described in the plant specific submittal(s) and the associated safety evaluation [Reference 48] and determined Exelon was in compliance with NRC Order EA-12-049 [Reference 45]. Since the original IPEEE, a re-evaluation was performed for external flooding that concluded external floods do not pose a significant threat to plant safety because the exceedance frequency of the design basis flood is less than 10^{-5} per year [Reference 47].

The major concern in a high-wind or tornado scenario are the wind loads imposed on the buildings/major structures and the potential for wind-generated missiles to disable systems or components necessary to shut down the plant or maintain the plant in a safe shutdown condition. GNPP wind and tornado loadings are evaluated under Section 3.3 of UFSAR [Reference 39]. All Class I buildings and structures are capable of withstanding tornado winds corresponding to 300 mph tangential velocities and a differential pressure drop of 3 psi with no loss of function. In addition, all Class I buildings and structures were also designed to withstand various postulated tornado generated missiles, including a steel rod, 1-in diameter and 3-ft long, weighing 8 lb, and a wooden utility pole, 13.5-in diameter and 35-ft long, weighing 1490 lb [Reference 39]. All DBNPS FLEX equipment is stored in structures with designs that are robust such that no one external event can reasonably fail the FLEX capability [Reference 45]. There have been no major changes to the buildings/major structures or location of important safety equipment within them since the IPEEE submittals that negatively impact plant vulnerability to external events. The only significant changes improve Ginna's ability to respond to external hazards and decrease overall plant risk: modifications made to the Standby Auxiliary Feedwater (SAFW) system to have independent power supplies (diesel generators), new water storage tank for SAFW (redundant to service water) which was built to be tornado-proof, alternate RCS injection pump for Small LOCA makeup (redundant and separate location from the normal charging pumps), installation of new Westinghouse RCP shutdown seals, and the addition of FLEX equipment and procedure additions or changes which provide the station with additional response capability to an event. Therefore, it is concluded that no new factors have been introduced at GNPP since the submittals of the IPEEE that would result in an increase in the risk associated with high winds, tornadoes, or tornado missiles.

No significant changes have been made that would affect the IPEEE evaluations of highway transportation, railroads, waterways, pipelines, military facilities, or industrial facilities. This evaluation is maintained in Section 2.2 of the UFSAR [Reference 39]. According to the Federal Aviation Administration's Air Traffic Activity System, air traffic at the Greater Rochester International Airport, the closest airport serving commercial airlines, has significantly decreased since the time of the IPEEE. Based on the information summarized here from the IPEEE [References 44, 46, and 47] and maintained in the UFSAR [Reference 39], these hazards are screened from this analysis.

5.2.8 Defense-In-Depth Impact

Regulatory Guide 1.174, Revision 3 [Reference 4] describes an approach that is acceptable for developing risk-informed applications for a licensing basis change that considers engineering and applies risk insights. One of the considerations included in RG 1.174 is Defense in Depth. Defense in Depth is a safety philosophy that employs successive compensatory measures to provide accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The following seven considerations will serve to evaluate the proposed licensing basis change for overall impact on Defense in Depth.

1. <u>Preserve a reasonable balance among the layers of defense.</u>

The use of the risk metrics of LERF, population dose, and conditional containment failure probability collectively ensures the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. The change in LERF is "small" per RG 1.174, and the change in population dose and CCFP are "small" as defined in this analysis and consistent with NEI 94-01 Revision 3-A.

2. <u>Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.</u>

The adequacy of the design feature (the containment boundary subject to Type A testing) is preserved as evidenced by the overall "small" change in risk associated with the Type A test frequency change.

3. Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.

The redundancy, independence, and diversity of the containment subject to the Type A test is preserved, commensurate with the expected frequency and consequences of challenges to the system, as evidenced by the overall "small" change in risk associated with the Type A test frequency change.

4. Preserve adequate defense against potential CCFs.

Adequate defense against CCFs is preserved. The Type A test detects problems in the containment which may or may not be the result of a CCF; such a CCF may affect failure of another portion of containment (i.e., local penetrations) due to the same phenomena. Adequate defense against CCFs is preserved via the continued performance of the Type B and C tests and the performance of inspections. The change to the Type A test, which bounds the risk associated with containment failure modes including those involving CCFs, does not degrade adequate defense as evidenced by the overall "small" change in risk associated with the Type A test frequency change.

5. Maintain multiple fission product barriers.

Multiple Fission Product barriers are maintained. The portion of the containment affected by the Type A test extension is still maintained as an independent fission product barrier, albeit with a "small" change in the reliability of the barrier.

6. Preserve sufficient defense against human errors.

Sufficient defense against human errors is preserved. The probability of a human error to operate the plant, or to respond to off-normal conditions and accidents is not significantly affected by the change to the Type A testing frequency. Errors committed during test and maintenance may be reduced by the less frequent performance of the Type A test (less opportunity for errors to occur).

7. Continue to meet the intent of the plant's design criteria.

The intent of the plant's design criteria continues to be met. The extension of the Type A test does not change the configuration of the plant or the way the plant is operated.

5.3 Sensitivities

5.3.1 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 5-18 – Table 5-20. The steel liner corrosion likelihood was increased by a factor of 1000, 10000, and 100000. Except for extreme factors of 10000 and 100000, which are extremely unlikely because they are many orders of magnitude larger than the corrosion factor calculated in Section 5.2.6, the corrosion likelihood is relatively insensitive to the results.

Table 5-18 – 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)
Corrosion Likelihood X 1	2.38E-08	7.93E-08	1.19E-07	5.55E-08	9.52E-08	3.97E-08
Corrosion Likelihood X 1000	2.40E-08	8.34E-08	1.33E-07	5.94E-08	1.09E-07	4.99E-08
Corrosion Likelihood X 10000	2.59E-08	1.20E-07	2.63E-07	9.44E-08	2.37E-07	1.42E-07
Corrosion Likelihood X 100000	4.50E-08	4.90E-07	1.56E-06	4.45E-07	1.51E-06	1.07E-06

Table 5-19 –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)
Corrosion Likelihood X 1	3.80E-01	3.85E-01	3.89E-01	5.14E-03	8.81E-03	3.67E-03
Corrosion Likelihood X 1000	3.80E-01	3.85E-01	3.89E-01	5.18E-03	8.88E-03	3.70E-03
Corrosion Likelihood X 10000	3.80E-01	3.86E-01	3.90E-01	5.59E-03	9.59E-03	4.00E-03
Corrosion Likelihood X 100000	3.82E-01	3.92E-01	3.99E-01	9.71E-03	1.66E-02	6.93E-03

Table 5-20 –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)
Corrosion Likelihood X 1	7.30E-02	2.43E-01	3.65E-01	1.70E-01	2.92E-01	1.22E-01
Corrosion Likelihood X 1000	7.36E-02	2.45E-01	3.68E-01	1.72E-01	2.95E-01	1.23E-01
Corrosion Likelihood X 10000	7.95E-02	2.65E-01	3.97E-01	1.85E-01	3.18E-01	1.32E-01
Corrosion Likelihood X 100000	1.38E-01	4.60E-01	6.90E-01	3.22E-01	5.52E-01	2.30E-01

5.3.2 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 [Reference 24]. 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 5-21 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 5-21 – GNPP 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 Table 5-10 and Table 5-11 for the expert elicitation sensitivity yields the results in Table 5-22.

Table 5-22 – GNPP 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	6.72E-06	6.68E-06	2.27E+04	1.52E-01	6.58E-06	1.49E-01	6.51E-06	1.48E-01
2	2.66E-09	2.66E-09	3.38E+06	8.98E-03	2.66E-09	8.98E-03	2.66E-09	8.98E-03
3a	N/A	4.02E-08	2.27E+05	9.14E-03	1.34E-07	3.05E-02	2.01E-07	4.57E-02
3b	N/A	2.56E-09	2.27E+06	5.82E-03	8.54E-09	1.94E-02	1.28E-08	2.91E-02
7	3.66E-06	3.66E-06	9.61E+05	3.52E+00	3.66E-06	3.52E+00	3.66E-06	3.52E+00
8	4.17E-07	4.17E-07	4.89E+06	2.04E+00	4.17E-07	2.04E+00	4.17E-07	2.04E+00
Totals	1.08E-05	1.08E-05	1.18E+07	5.74E+00	1.08E-05	5.77E+00	1.08E-05	5.80E+00
ΔLERF (3 per 10 yrs base)	N/A				5.98E-09		1.02E-08	
ΔLERF (1 per 10 yrs base)	N/A				N/A		4.27E-09	
CCFP	37.82%				37.88%		37.92%	
ΔCCFP (3 per 10 yrs base)	N/A				0.06%		0.09%	
ΔCCFP (1 per 10 yrs base)	N/A				N/A		0.04%	

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 GNPP 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	
		CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year	CDF/Year	Person-Rem/Year
1	2.27E+04	6.60E-06	1.50E-01	6.32E-06	1.44E-01	6.12E-06	1.39E-01
2	3.38E+06	2.66E-09	8.98E-03	2.66E-09	8.98E-03	2.66E-09	8.98E-03
3a	2.27E+05	9.56E-08	2.17E-02	3.19E-07	7.23E-02	4.78E-07	1.08E-01
3b	2.27E+06	2.38E-08	5.40E-02	7.93E-08	1.80E-01	1.19E-07	2.70E-01
7	9.61E+05	3.66E-06	3.52E+00	3.66E-06	3.52E+00	3.66E-06	3.52E+00
8	4.89E+06	4.17E-07	2.04E+00	4.17E-07	2.04E+00	4.17E-07	2.04E+00
Total		**1.08E-05**	**5.80E+00**	**1.08E-05**	**5.97E+00**	**1.08E-05**	**6.09E+00**
ILRT Dose Rate from 3a and 3b							
ΔTotal Dose Rate	From 3 Years	N/A		1.70E-01		2.92E-01	
	From 10 Years	N/A		N/A		1.22E-01	
%ΔDose Rate	From 3 Years	N/A		2.94%		5.03%	
	From 10 Years	N/A		N/A		2.04%	
3b Frequency (LERF)							
ΔLERF	From 3 Years	N/A		5.55E-08		9.52E-08	
	From 10 Years	N/A		N/A		3.96E-08	
CCFP %							
ΔCCFP%	From 3 Years	N/A		0.514%		0.881%	
	From 10 Years	N/A		N/A		0.367%	

These results are summarized and compared to the Regulatory Guide 1.174 acceptance guidelines [Reference 4] in Section 7.0.

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 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 9.52E-8/year using the EPRI guidance. Therefore, the estimated change in LERF is determined to be "very small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4]. The risk change resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years bounds the 1 in 10 years to 1 in 15 years risk change. Considering the increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 10 years to 1 in 15 years is estimated as 3.96E-8, the risk increase is "very small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4].
- When external event risk is included, the increase in 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 3.91E-7/year using the EPRI guidance, and total LERF is 1.61E-6/year. As such, the estimated change in LERF is determined to be "small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4]. The risk change resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years bounds the 1 in 10 years to 1 in 15 years risk change. When external event risk is included, the increase in LERF resulting from a change in the Type A ILRT test interval from 1 in 10 years to 1 in 15 years is estimated as 2.28E-7 and the total LERF is 1.45E-6. Therefore, the risk increase is "small" using the acceptance guidelines of Regulatory Guide 1.174 [Reference 4]. As discussed in Sections 5.1.3 and 5.2.7, the EPRI methodology used to estimate the increase in LERF and the models used to estimate total LERF are conservative. Therefore, the conservative methodology adds margin.
- 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 0.29 person-rem/year. NEI 94-01 [Reference 1] states that a 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 0.881%. NEI 94-01 [Reference 1] states that increases in CCFP of ≤ 1.5% is small. Therefore, this increase is judged to be small.

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

<u>Previous Assessments</u>

Historical ILRT extension evaluations provide further corroboration to support the conclusion that increasing the ILRT interval has only a small impact on plant risk. 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 GNPP confirm these general conclusions on a plant-specific basis considering the severe accidents evaluated for GNPP, the GNPP containment failure modes, and the local population surrounding GNPP.

A. ATTACHMENT 1

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

The GN116A version of the Ginna PRA model is the most recent evaluation of internal event risks. The Ginna PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the Ginna PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

Exelon Generation Company, LLC (Exelon) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for all operating Exelon nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process and the use of self-assessments and independent peer reviews. Prior to joining the Exelon nuclear fleet in 2014, comparable practices were in place when Ginna was owned and operated by Constellation Energy Nuclear Group (CENG). Because of the similarities between the CENG and Exelon practices, no additional discussion specifically regarding the legacy CENG approach will be provided. The following information describes the Exelon approach (and by extension the CENG approach) to PRA model maintenance, as it applies to the Ginna PRA.

A.1.1 PRA Maintenance and Update

The Exelon risk management process ensures that the applicable PRA model is an accurate reflection of the as-built and as-operated plants. This process is defined in the Exelon Risk Management program, which consists of a governing procedure (ER-AA-600, "Risk Management") and subordinate implementation training and reference materials (T&RM's).

- Exelon T&RM ER-AA-600-1015, "Full Power Internal Event (FPIE) PRA Model Update," delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Exelon nuclear generation sites.
- ER-AA-600-1061 "Fire PRA Model Update and Control" delineates the responsibilities and guidelines for updating the station fire PRA.

The overall Exelon Risk Management program, including ER-AA-600-1015 and ER-AA-6001061, defines the process for implementing regularly scheduled and interim PRA model updates, tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.), and controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events PRA models for Exelon nuclear generation sites.
- As an NFPA 805 plant, all modifications are reviewed to ensure the modification meets the fire requirements during the initial design phase of a modification per ER-AA-600-1068 [Reference 60].
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and

modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65(a)(4)).

As indicated previously, RG 1.200 also requires that additional information be provided as part of the LAR submittal to demonstrate the technical adequacy of the PRA model used for the risk assessment. Each of these items (plant changes not yet incorporated in to the PRA model, relevant peer review findings, consistency with applicable PRA Standards, and the identification of key assumptions) will be discussed in turn.

A.1.2 Plant Changes Not Yet Incorporated into the PRA Model

Each Exelon station maintains an updating requirements evaluation (URE) database to track all enhancements, corrections, and unincorporated plant changes. During the normal screening conducted as part of the plant change process, if a potential model update is identified, a new URE database item is created. Depending on the potential impact of the identified change, the requirements for incorporation will vary.

As part of this PRA evaluation, a review of open items in the URE database for Ginna is performed, and an assessment of the impact on the results of the application is made. Some UREs may affect the LERF containment modeling. Open URE 834 pertains to F&O LE-C10-01, which states credit for scrubbing was not taken. Per the response to RAI 17 for the TSTF-425 LAR [Reference 58], scrubbing may be applicable to the following three containment bypass conditions: 1) a steam generator tube rupture event with feedwater available, or 2) internal flood scenarios with an interfacing system LOCA and the affected auxiliary building room flooded, or 3) sequences where the interfacing system LOCA break is in the RHR pits. Since these are only Class 8 (SGTR or ISLOCA) sequences, there would be no effect on ILRT ΔLERF (change in Class 3b risk).

Open URE 837 pertains to LERF quantification, which is used in this analysis, and tracks finding LE-C9a, which is capability category I. Since NEI 94-01 endorses using PRA models conformed to capability category I of the ASME/ANS standard, the Ginna PRA model is of sufficient quality to use for this ILRT analysis. Additionally, this IRLT extension analysis has significant margin to the Regulatory Guide 1.174 acceptance guidelines [Reference 4], so any model update from this URE is judged to be sufficiently small so as to not affect this IRLT extension analysis.

Additional open UREs may also affect overall CDF and LERF quantification results, which are used to calculate change in risk metrics for the ILRT extension evaluation. After evaluating all open UREs for their effect on CDF and LERF, it is concluded the aggregate of open UREs leads to the current PRA model being conservative. A conservative CDF alone would lead to conservative calculations for the ILRT extension; since LERF is subtracted from CDF to calculate the risk increase due to the ILRT extension, if the LERF conservatism is greater than the CDF conservatism, the ILRT extension calculations would not be conservative. Since the magnitude of CDF and LERF conservatisms are unknown, a sensitivity is performed where LERF is not subtracted from CDF when calculating the change in risk for the ILRT extension (as described in Section 5.2.1). Sensitivity results are shown in Table A-1 through Table A-4.

Table A-1 – Sensitivity: 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)	2.48E-08	8.26E-08	1.24E-07
ΔLERF (3 year baseline)		5.78E-08	9.91E-08
ΔLERF (10 year baseline)			4.13E-08

The increase in the overall probability of LERF due to Class 3b sequences is less than 10^{-7}. Therefore, the ΔLERF is considered very small [Reference 4].

NEI 94-01 [Reference 1] states that a 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 A-2, the results of this calculation meet the dose rate criteria.

Table A-2 – Sensitivity: Impact on Dose Rate due to Extended Type A Testing Intervals

ILRT Inspection Interval	10 Years	15 Years
ΔDose Rate (3 year baseline)	1.774E-01	3.042E-01
ΔDose Rate (10 year baseline)		1.267E-01
%ΔDose Rate (3 year baseline)	3.058%	5.242%
%ΔDose Rate (10 year baseline)		2.119%

Table A-3 – Sensitivity: Impact on CCFP due to Extended Type A Testing Intervals

ILRT Inspection Interval	3 Years (baseline)	10 Years	15 Years
f(ncf) (/yr)	6.70E-06	6.64E-06	6.60E-06
f(ncf)/CDF	0.620	0.614	0.611
CCFP	0.380	0.386	0.389
ΔCCFP (3 year baseline)		0.535%	0.917%
ΔCCFP (10 year baseline)			0.382%

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 0.917%. Therefore, this increase is judged to be small.

This sensitivity methodology is also applied to the external event risk ILRT extension calculations.

Table A-4 – Sensitivity: GNPP 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	7.59E-08	2.53E-07	3.79E-07	3.03E-07
Internal Events	2.48E-08	8.26E-08	1.24E-07	9.91E-08
Combined	1.01E-07	3.35E-07	5.03E-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 due to increasing the ILRT interval from 3 to 15 years is 4.03E-7, which meets the guidance for small change in risk, as it exceeds 1.0E-7/yr and remains less than a 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 values are calculated below:

$$\text{LERF} = \text{LERF}_{\text{internal}} + \text{LERF}_{\text{fire}} + \text{LERF}_{\text{seismic}} + \text{LERF}_{\text{class3Bincrease}}$$
$$\text{LERF}_{15\text{yr}} = 4.34\text{E-}7/\text{yr} + 6.31\text{E-}7/\text{yr} + 1.56\text{E-}7/\text{yr} + 4.03\text{E-}7/\text{yr} = 1.62\text{E-}6/\text{yr}$$

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.

A.2. Applicability of Peer Review Findings and Observations (F&Os)

The technical acceptability of the Ginna PRA models has been demonstrated by the peer review process. The purpose of the industry PRA peer review process is to provide a method for establishing the technical capability and adequacy of a PRA relative to expectations of knowledgeable practitioners, using a set of guidance that establishes a set of minimum requirements. PRA peer reviews continue to be performed as PRAs are updated (and upgraded) to ensure the ability to support risk-informed applications and has proven to be a valuable process for establishing technical adequacy of nuclear power plant PRAs.

There have been three relevant peer reviews conducted on the current PRA model.

- The 2009 peer review for the PRA ASME model update identified 309 Supporting Requirements (SR) applicable to the Ginna PRA. Of these 29 were not met, 2 met capability category (CC) 1, 13 met CC 1/2, 31 met CC 2, 22 met CC 2/3, 14 met CC 3, and 198 fully met all capability requirements. There were 24 findings and observations (F&Os) issued to address the identified gaps to compliance with the PRA standard. Subsequent to the peer review, 13 of the findings have been addressed and 11 are still open pending the next model update. The open F&Os are listed in Table A-5, which includes what, if any, impact there may be to the ILRT extension.
- The 2012 fire PRA peer review for the PRA ASME model update identified 183 Supporting Requirements (SR) to be reviewed for the Ginna PRA. Of these 2 were not met, 2 met capability category (CC) 1, 8 met CC 1/2, 17 met CC 2, 13 met CC 2/3, 7 met CC 3, and 118 fully met all capability requirements and 16 were not applicable. There were 19 findings and 22 suggestions issued to address potential gaps to compliance with the PRA standard. There were 3 Best Practices. All the findings that impact the fire PRA were closed prior to the initial NFPA 805 submittal. As the results of this peer review have already been communicated to the NRC as part of the NFPA-805 submittal [Reference 57] and subsequent requests for additional information (RAI), these will not be catalogued in this document.
- A peer review was conducted to assess actions taken to address existing finding-level F&Os [Reference 59]. The June 2017 FPIE review performed an independent assessment of finding-level F&Os from previous peer reviews. Finding-level F&Os that were reviewed and were determined to have been adequately addressed through this technical review are considered “closed.” These closed F&Os are no longer relevant to the current PRA model. The technical review team determined that 17 of the 23 finding-level F&Os were resolved. Four of the finding-level F&Os remain open. The remaining two finding-level F&Os were partially resolved but require further documentation (i.e., all technical aspects were resolved) [Reference 59].

The remaining gaps are documented in the URE database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

Table A-5 Ginna FPIE PRA Focused-Scope Peer Review Facts & Observations

SR	Topic	Status	Finding/Observation	Disposition	ILRT Impact
SC-A2	SPECIFY the plant parameters (e.g., highest node temperature, core collapsed liquid level) and associated acceptance criteria (e.g., temperature limit) to be used in determining core damage. Select these parameters such that determination of core damage is as realistic as practical, in a manner - consistent with current best practice. DEFINE computer code-predicted acceptance criteria with sufficient margin on the code-calculated values to allow for limitations of the code, sophistication of the models, and uncertainties in the results, in a manner consistent with the requirements specified under HLR-SC-B. Examples of measures for core damage suitable for Capability Category II/III, that have been used in PRAs, include (a) collapsed liquid level less than 1?3 core height or code-predicted peak core temperature >2,500°F (BWR) (b) collapsed liquid level below top of active fuel for a prolonged period, or code-pre-dieted core peak node temperature >2,200°F using a code with detailed core modeling; or code-predicted core peak node temperature >1,800°F using a code with simplified (e.g., single-node core model, lumped para- meter) core modeling; or code-predicted core exit temperature >1,200°F for 30 min using a code with simplified core modeling (PWR).	Open Met-CC I	F&O SC-A2-01: The definition of core damage documented in the Ginna-AS- Notebook-Rev-1 Section 2.2 is consistent with the examples of measures for core damage suitable for Capability Category I as defined in NUREG/CR-4550. For Category II Ginna could use the code-predicted core exit temperature >1,200°F for 30 min using PCTRAN (code with simplified core modeling (PWR)).	We agree with the peer reviewers that the approach taken in the Ginna PRA is overly conservative and not consistent with the requirements of Category II. The peer reviewers suggested using a core exit temperature of 1200°F for 30 minutes as the criterion for core damage, but we would recommend using either that criterion or a peak core node temperature of 1800"F. Based on a review of the PCTRAN results, it is likely that the 1800"F peak core temperature would be reached earlier than the time at which the core exit temperature would be greater than 1200°F for 30 minutes.	Over the typical complete loss of decay heat removal timing success criteria, the delta time between core uncovery and CET temperatures reach 1200°F for 30 minutes or 1800° peak center line is fairly small. As such, the timing benefit is not expected to be large except for the fast moving events such as large break LOCAs. For these events, we use the UFSAR success criteria. Although this is not expected to be a significant effect, the model remains a conservative CAT I. Therefore, the model used for ILRT extension analysis may be slightly conservative. Since this IRLT extension analysis has significant margin to the Regulatory Guide 1.174 acceptance guidelines [Reference 4], any model update from this finding is judged to be sufficiently small so as to not affect this IRLT extension analysis.

Table A-5 Ginna FPIE PRA Focused-Scope Peer Review Facts & Observations

SR	Topic	Status	Finding/Observation	Disposition	ILRT Impact
LE-C11 [2005: LE-C9a]	JUSTIFY any credit given for equipment survivability or human actions that could be impacted by containment failure,	CAT I	F&0 LE-C9a-01: It does not appear that credit was taken for continued operation of equipment and operator actions that could be impacted by containment failure. This is a requirement of the standard to move from Category I to Category II.	The requirement is to justify credit taken for equipment survivability or human actions that could be affected by containment failure. Since no such credit was taken, the SR should have been judged as not applicable (N/A). This is analogous to the assessment of LE-C7 (old LE-C6) which was judged by the peer reviewers as N/A because human actions that support the accident progression analysis were not credited. Also, note that, in the Calvert Cliffs peer review, the peer reviewers judged this SR as N/A for the same reason. Only if post-containment failure equipment operations or human actions are modeled in the future would it be necessary to provide engineering analysis and written justification as part of the PRA documentation. Otherwise, no additional work is needed.	As no equipment or HRA is credited post-containment failure, the PRA model remains a conservative CAT I. See Section A.1.2 for details.
LE-C13 [2005: LE-C10]	PERFORM a containment bypass analysis in a realistic manner. JUSTIFY any credit taken for scrubbing (i.e., provide an engineering basis for the decontamination factor used).	CAT I	F&0 LE-C10-01: Credit for scrubbing was not taken. A sensitivity for impact of scrubbing was performed and it was determined that the impact of not considering scrubbing is negligible. This is a requirement of the standard to move from Category I to Category II.	Review the possible credit for release scrubbing to reduce LERF.	Negligible impact to the ILRT extension analysis. A sensitivity for impact of scrubbing was performed and it was determined that the impact of not considering scrubbing

Table A-5 Ginna FPIE PRA Focused-Scope Peer Review Facts & Observations

SR	Topic	Status	Finding/Observation	Disposition	ILRT Impact
					is negligible. Since this change would only affect Class 8 (SGTR or ISLOCA) sequences [Reference 58], there would be no effect on ILRT ΔLERF (change in Class 3b risk).
IFSN-A6 [2005: IF-C3]	For the SSCs identified in iFSN-A5 (2005 text: IF-C2c), IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. EITHER: a) ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions; OR b) NOTE that these mechanisms are not included in the scope of the evaluation.	Open	F&O IF-C3-01: There is no discussion of failures due to jet impingement or pipe whip. There is limited consideration of failure due to humidity/high temperature due to failure of heating steam lines. There is also no discussion of criteria employed to consider the potential for spray failures. To meet Capability Category II, it is necessary either to provide at least a qualitative assessment of the potential for jet impingement and pipe whip, or to state that these failure mechanisms were not considered. It is also required that potential spray failures be evaluated. While spray failures are discussed, there are no criteria specified that would provide assurance that they had been considered in a consistent and adequately comprehensive manner. Provide the requisite discussion of pipe whip and jet impingement to satisfy the standard. Specify appropriate criteria for spray impacts, and assure that the potential spray failures adequately reflect these criteria.	Cat II: INCLUDE failure by submergence and spray in the identification process. ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions. [SAIC note: these mechanisms include submergence, spray, jet impingement, pipe whip, humidity, condensation, temperature concerns] Revise the Internal Flooding Study (51- 9100978-000) to describe the criteria used to determine the potential for failure resulting from spray. Reference Appendix C for a listing of components impacted by spray. Describe how potential spray impact was addressed in the model. Confirm that the assignment	Failures due to jet impingement and pipe whip are now discussed in Section 3.3.1 of the Internal Flood Notebook G1-IF-0000 r1. Failures due to Spray are discussed in Section 3.3.2. Impacts due to spray were assumed to exist within 10 feet of a break location. Spray events are discussed in the IF Flood notebook Section 4.5. Two locations were identified in the Aux Building where Fire Service Water could impact safety-related busses and these are explicitly modeled (FL-ABM-FSWBUS15 and FL-ABO-FSW-BUS14). URE 1179 documents that IF Notebook needs Appendix C completed to complete documentation of

Table A-5 Ginna FPIE PRA Focused-Scope Peer Review Facts & Observations

SR	Topic	Status	Finding/Observation	Disposition	ILRT Impact
				of spray impact is consistent with the criteria used. In addition, include a qualitative discussion of the potential impact of jet impingement, pipe whip, humidity, condensation, and temperature effects.	spray impacts and modeling of additional spray floods if appropriate. This is expected to have a negligible impact on the ILRT extension analysis.
IFQU-B1 [2005: IF-F1]	DOCUMENT the internal flood accident sequences and quantification in a manner that facilitates PRA applications, upgrades, and peer review.	Open	F&O IF-F1-01: The documentation is comprised primarily of the internal flooding notebook, supplemented heavily with information provided in a set of Excel worksheets. The notebook is annotated to provide a link to elements of the worksheets, and an "assumption" provides the formal tie between the notebook and the worksheets. Some areas in which the links were indirect or missing were noted. In general, the manner in which important parts of the flood analysis are documented in what would usually be characterized as an informal set of worksheets is judged not to meet the requirement that the analysis be documented in a manner that facilitates applications, upgrades, and peer review. In addition to developing a single integrated set of documentation for the internal flood analysis, there were several areas in which additional documentation would make the analysis more tractable have been provided in connection to other SRs. These include the following: ▪ Include a set of simplified arrangement drawings to explicate the definition of flood areas and help in understanding aspects such as flood propagation. ▪ Tabulate the flood areas and identify clearly which are screened and which retained for further analysis to make the	Documentation only: Revise the Internal Flooding Study (51 -9100978 - 000) to meet the documentation requirements of the 2009 Standard. Address NRC Resolutions as appropriate. It is recommended that the Study be reformatted to be consistent with the HLRs and SRs of the Standard, integrating appropriate parts of the worksheets into the primary document. This will provide a document that can be easily reviewed against the standard and easily followed by personnel not involved in the original analysis. Consistent with the F&O, include the following in the revised Study: ▪ Define the criteria used to determine whether a PRA component was susceptible to failure due to spray. As stated in the F&O closure report [Reference	This documentation item will not impact the ILRT extension analysis. This item has largely been addressed by adding tables in Section 5.2 that show the development of each initiating event frequency, adding an Initiating Event Summary Table (section 5.2.17), adding a simplified set of arrangement drawings showing each flood area (Appendix K), defining spray modeling criteria (Section 3.3.2) and identifying for each flood area whether it was screened and the screening criterion used (Table 4.6).

Table A-5 Ginna FPIE PRA Focused-Scope Peer Review Facts & Observations

SR	Topic	Status	Finding/Observation	Disposition	ILRT Impact
			process more tractable. Specify clearly which criteria (qualitative or quantitative) are employed in screening each flood area. ▪ Define explicitly the criteria used to perform quantitative screening as noted in Section 6.0. ▪ Define the criteria used to determine whether a PRA component was susceptible to failure due to spray.	59], the rest of the items have been addresssed.	
HR-G3	When estimating HEPs EVALUATE the impact of the following plant-specific and scenario-specific performance shaping factors: (a) quality [type (classroom or simulator) and frequency] of the operator training or experience (b) quality of the written procedures and administrative controls (c) availability of instrumentation needed to take corrective actions (d) degree of clarity of the cues/indications (e) human-machine interface (f) time available and time required to complete the response (g) complexity of the required response (h) environment (e.g., lighting, heat, radiation) under which the operator is working (i) accessibility of the equipment requiring manipulation (j) necessity, adequacy, and availability of special tools, parts, clothing, etc.	Partially Resolved	F&O HR-G3-01: Details regarding certain elements of the analysis were lacking in the HRA Calculator for a sufficient number of HFEs to judge that this requirement was not met. Evidence that the relevant aspects cited in the SR are addressed for each HFE is critical to assuring that an appropriate analysis has been performed. This is particularly important in the case of HRA, for which the methods are less straightforward than they are for many other parts of the PRA.	Issue: In item (d) of CC II, III, clarify that 'clarity' refers to the meaning of the cues, etc. In item (g) of CC II, III, clarify that complexity refers to both determining the need for and executing the required response. Resolution: Cat I, II, and III: (d) degree of clarity of the meaning of cues / indications (g) complexity of detection, diagnosis and decision-making, and executing the required response.	No impact to the ILRT extension analysis. The HRAs have been reviewed to ensure the needed parameters for the evaluation have been populated. CBDM is now used as a max function of CBDT and HCR/ORE. RCHFDMAKEUP as a specific example has a timing basis from Key Input 51. When the annunciator model is used, there is a clear discussion as to the applicability.

A.2.1 Seismic PRA

The GNPP IPEEE seismic risk analysis did not quantify a CDF impact. The SCDF calculation is summarized in Section 5.2.7 and detailed in Appendix B.

A.3. Consistency with Applicable PRA Standards

Based on the peer reviews, independent assessment of F&O resolutions, and the focused scope peer reviews, it is concluded that the current Ginna internal events and fire PRA models mostly conform to capability category II of ASME RA-Sb-2009, ASME/ANS Standard for Probabilistic Risk Assessment of Nuclear Power Plant Applications as endorsed by RG 1.200 Revision 2 (with the remaining few items conforming to capability category I of ASME RA-Sb-2009). Since NEI 94-01 endorses using PRA models conformed to capability category I of the ASME/ANS standard, using these models for this ILRT analysis meets technical adequacy requirements.

B. ESTIMATED SEISMIC CDF CALCULATION

The seismic hazard input is obtained from Reference 51 and shown in Table B-1. Several points have been interpolated to provide values at convenient seismic hazard points. The mean fractile occurrence frequencies of Table B-1 are used in the calculations here; use of mean values is a typical and expected PRA practice. Table B-2 shows the seismic hazard intervals used in this analysis along with their representative PGA (used for fragility calculation) and occurrence frequency. Nine hazard intervals are used in this analysis; this is consistent with the number of hazard intervals used in industry SPRAs [Reference 55].

Table B-6 – Steel Liner Corrosion Base Case
(Reproduced from Reference 51 Table A-1a. Mean and Fractile Seismic Hazard Curves for PGA at Limerick)

AMPS (g)	Mean	0.05	0.16	0.5	0.84	0.95	Notes
0.0005	5.050E-02	2.46E-02	3.95E-02	5.05E-02	6.26E-02	7.13E-02	
0.001	3.620E-02	1.38E-02	2.60E-02	3.57E-02	4.77E-02	5.66E-02	
0.005	8.150E-03	2.25E-03	4.25E-03	6.93E-03	1.10E-02	1.98E-02	
0.01	3.200E-03	8.60E-04	1.36E-03	2.49E-03	4.19E-03	1.01E-02	
0.015	1.710E-03	4.37E-04	6.45E-04	1.21E-03	2.19E-03	6.09E-03	
0.03	5.190E-04	1.02E-04	1.55E-04	3.01E-04	6.73E-04	2.19E-03	
0.05	2.070E-04	3.28E-05	5.27E-05	1.10E-04	2.80E-04	8.85E-04	
0.075	9.910E-05	1.44E-05	2.42E-05	5.20E-05	1.40E-04	4.01E-04	
0.08	9.102E-05						Interpolated Value[1]
0.1	5.870E-05	8.47E-06	1.44E-05	3.14E-05	8.35E-05	2.22E-04	
0.15	2.780E-05	3.95E-06	7.03E-06	1.60E-05	4.01E-05	9.51E-05	
0.25	1.409E-05						Interpolated Value[1]
0.3	7.240E-06	8.72E-07	1.77E-06	4.50E-06	1.10E-05	2.25E-05	
0.35	6.043E-06						Interpolated Value[1]
0.5	2.450E-06	2.22E-07	5.05E-07	1.51E-06	3.95E-06	7.55E-06	
0.7	1.252E-06						Interpolated Value[1]
0.75	9.520E-07	5.75E-08	1.53E-07	5.42E-07	1.60E-06	3.09E-06	
0.9	6.574E-07						Interpolated Value[1]
1	4.610E-07	1.84E-08	5.75E-08	2.42E-07	7.89E-07	1.60E-06	
1.3	2.750E-07						Interpolated Value[1]
1.5	1.510E-07	2.96E-09	1.16E-08	6.54E-08	2.60E-07	5.75E-07	
1.65	1.376E-07						Interpolated Value[1]
3	1.670E-08	1.20E-10	4.56E-10	4.25E-09	2.68E-08	7.34E-08	
5	2.440E-09	5.05E-11	9.11E-11	4.07E-10	3.37E-09	1.15E-08	
7.5	4.300E-10	3.47E-11	5.35E-11	9.79E-11	5.35E-10	2.07E-09	
10	1.120E-10	3.01E-11	4.01E-11	9.11E-11	1.62E-10	5.75E-10	

[1] Interpolated values (using straight line interpolation) performed for this analysis for use in calculation of the hazard interval frequencies. These specific PGA points are not listed in Table A-1a of Reference 51. Interpolations are performed only for mean values.

The representative PGA (used in the fragility calculations) for seismic hazard intervals 1 through 8 is defined using the geometric mean (the approach used by the NRC [Reference 56] as well as commonly used in industry SPRAs [Reference 55]) and calculated as the square root of the product of the PGA values at the beginning and end of each hazard interval. For the last seismic hazard interval 9, the representative PGA value is defined as 1.1 times the PGA at the beginning of the last interval since this interval has no upper limit; the precision of the selection of the representative PGA for the final hazard interval is not significant to the final results because the plant level HCLPF failure probability for the final interval is effectively 1.0.

The occurrence frequency for each seismic hazard interval (except for the final interval) is calculated as the exceedance frequency of the beginning point of the interval minus the exceedance frequency of the end point of the interval. The frequency of the final (highest) ground motion interval is the exceedance frequency at the beginning point of that interval. The portion of the GNPP seismic hazard curve below 0.08g (i.e., the GNPP operating basis earthquake, OBE [Reference 51]) is not a significant contribution to calculated risk and indeed the plant would likely remain on line and not trip. The portion of the hazard curve below the OBE is not included in this calculation; this is consistent with typical seismic risk calculations and the industry guidelines (e.g., Reference 55). The plant level HCLPF value for GNPP used in this analysis (i.e., 0.2g) is sufficiently higher than the OBE, and because the HCLPF represents the 1% likelihood of failure value (with respect to the mean hazard curve), there is no significant risk associated with the portion of the hazard curve below the OBE (it would contribute <<0.1% to the total calculated SCDF and SLERF if assumed to result in a plant trip and explicitly included in this calculation).

Table B-7 – Ginna Seismic Hazard Bins (Based on EPRI 2013 Hazard) (Mean, 1/yr)

ID	Seismic IE Interval Range (g, PGA)	Seismic IE Interval Representative Magnitude (g, PGA)	Hazard Interval Frequency
%G1	0.08 - 0.15	0.11	6.32E-05
%G2	0.15 - 0.25	0.19	1.37E-05
%G3	0.25 - 0.35	0.30	8.05E-06
%G4	0.35 - 0.5	0.42	3.59E-06
%G5	0.5 - 0.7	0.59	1.20E-06
%G6	0.7 - 1.0	0.84	7.91E-07
%G7	1.0 - 1.3	1.14	1.86E-07
%G8	1.3 - 1.5	1.40	1.24E-07
%G9	>1.5	1.65	1.51E-07

The seismic failure probability of the GNPP limiting HCLPF for each hazard interval is calculated using the following fragility equations. These are the typical lognormal fragility equations used in most hazard PRAs [Reference 55].

Fragility = Φ [ln(A/Am)/ βc],

where Φ = standard lognormal distribution function

A = g level

Am = median seismic capacity

The uncertainty parameters (beta values) are related as follows:

$$\beta c = \sqrt{\beta u^2 + \beta r^2}$$

HCLPF and Am are related as follows:

Am = HCLPF / $(e^{-1.65(\beta r + \beta u)})$

Seismic CDF is evaluated corresponding to the HCLPF value based on the GNPP IPEEE analysis, or 0.2g PGA. Values of βr and βu are calculated to obtain the desired βc = 0.4. Then the value of Am is calculated using the HCLPF value (0.2g).

With all parameters specified, the interval-specific failure probabilities are calculated as defined above. The interval-specific failure probabilities are shown in Table B-3 for each interval. Note that in Table B-3, the interval frequencies from Table B-2 are repeated for convenience.

The SCDF for each hazard interval is then the product of the interval frequency and the interval seismic failure probability. The total SCDF is the sum over all intervals of the interval SCDF. These results are shown in Table B-4, which also shows the percentage of total SCDF for each interval. As shown in Table B-4, the total estimated SCDF is 3.88E-6/yr.

Table B-8 – Seismic-Induced Failure Probabilities for Each Hazard Bin based on HCLPF from IPEEE

Seismic-Induced Failure	HCLPF	Am	βr	βu	βc	Seismic-Induced Failure Probability as a Function of Seismic Magnitude — Seismic Magnitude (g, PGA)								
						%G1	%G2	%G3	%G4	%G5	%G6	%G7	%G8	%G9
						0.11	0.19	0.30	0.42	0.59	0.84	1.14	1.40	1.65
IPEEE HCLPF	0.20	0.509	0.283	0.283	0.40	6.2E-5	7.9E-3	8.8E-2	3.1E-1	6.5E-1	8.9E-1	9.8E-1	9.9E-1	1.0E+0
Seismic IE Frequency (/yr)						6.32E-5	1.37E-5	8.05E-6	3.59E-6	1.20E-6	7.91E-7	1.86E-7	1.24E-7	1.51E-7

Table B-9 – Total Estimated SCDF and Contribution by Hazard Interval

	%G1	%G2	%G3	%G4	%G5	%G6	%G7	%G8	%G9	Total Limiting HCLPF SCDF (/yr)
SCDF Contribution per interval for Limiting HCLPF=0.2g PGA (/yr)	6.2E-5	7.9E-3	8.8E-2	3.1E-1	6.5E-1	8.9E-1	9.8E-1	9.9E-1	1.0E+0	3.9E-06
SCDF % Contribution per interval for Limiting HCLPF=0.2g PGA	0.1%	2.8%	18.2%	28.9%	20.0%	18.2%	4.7%	3.2%	3.9%	100.0%