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RS-17-051 10 CFR 50.90

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

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

References:
1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "Request for License Amendment to Revise Technical Specifications Section 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies," dated September 19, 2016

2. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "Withdrawal of Request for License Amendment to Revise Technical Specifications Section 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies," dated December 22, 2016

3. Letter from K. J. Green (U.S. NRC) to B. C. Hanson (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 – Withdrawal of Requested Licensing Action to Revise Technical Specification 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies Submitted to NRC for Acceptance Review (CAC Nos. MF8387 and MF8388)," dated December 23, 2016

In Reference 1, Exelon Generation Company, LLC (EGC) submitted a license amendment request for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively, to allow for the permanent extension of the Type A Integrated Leak Rate Testing (ILRT) and Type C Leak Rate Testing frequencies. However, EGC withdrew the proposed license amendment request in Reference 2 based on concerns that were identified during the NRC's acceptance review of the license amendment request. The NRC concerns were summarized and provided to EGC in Reference 3. Specifically, Reference 3 states that the Reference 1 application did not provide the following technical information in sufficient detail to enable the NRC to complete its detailed review, and this information should be included if EGC decides to resubmit the request:

- The results of a peer review of the internal events probabilistic risk assessment (PRA) that was conducted against all the supporting requirements of the PRA Standard ASME/ANS RA-Sa-2009, and Regulatory Guide 1.200, Revision 2, "An approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014), that were affected by any PRA upgrades;

- A list of the facts and observations (F&Os) from this peer review, with details of their disposition; and

- For any open or unresolved F&Os, justification for why not meeting the corresponding Capability Category I requirements has no impact on the requested licensing action.

As discussed in Attachment 3, EGC recently completed an independent peer review of the QCNPS internal events PRA model, and has dispositioned the findings related to F&Os, to address the concerns listed above.

Therefore, in accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," EGC requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for QCNPS, Units 1 and 2, respectively. The proposed change revises Technical Specifications (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for the permanent extension of the Type A Integrated Leak Rate Testing (ILRT) and Type C Leak Rate Testing frequencies.

Specifically, the proposed change revises QCNPS TS 5.5.12 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 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A, as the documents used by QCNPS to implement the performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J. This license amendment request also proposes an administrative change to TS 5.5.12 to delete references to Type A tests that have already occurred.

The proposed change is risk-informed and follows the guidance in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2. EGC has performed a QCNPS-specific evaluation to assess the risk impact of the proposed change. A copy of the risk assessment is provided in Attachment 3.

This request is subdivided as follows.

- Attachment 1 provides a description and evaluation of the proposed change.
- Attachment 2 provides a markup of the affected TS pages.
- Attachment 3 provides QC-LAR-03, "Risk Assessment for QCNPS Regarding the ILRT (Type A) Permanent Extension Request," Revision 2.

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

EGC requests approval of the proposed change by September 19, 2017, to support the extension of the Unit 2 ILRT, which is required to be performed during the outage in the spring of 2018. 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 Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 27th day of April 2017.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:
1. Evaluation of Proposed Change
2. Markup of Proposed Technical Specifications Pages
3. QC-LAR-03, "Risk Assessment for QCNPS Regarding the ILRT (Type A) Permanent Extension Request"

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Change

1.0 SUMMARY DESCRIPTION

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 Renewed Facility Operating License Nos. DPR-29 and DPR-30, for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2.

The proposed change revises Technical Specifications (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow the following:

- Increase the existing Type A integrated leakage rate test (ILRT) program test interval from 10 years to 15 years in accordance with Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A (Reference 2), and the conditions and limitations specified in NEI 94-01, Revision 2-A (Reference 8).

  Note: This change would make permanent, a test interval extension of the Type A, Appendix J ILRT testing of QCNPS Units 1 and 2, previously approved on March 8, 2004, in License Amendments No. 220 (Unit 1) and No. 214 (Unit 2) (Reference 17). These amendments provided a one-time TS change extending the Type A, Appendix J test interval from 10 to 15 years as applied to QCNPS, Units 1 and 2.

- Adopt an extension of the containment isolation valve (CIV) leakage rate testing (Type C) frequency from the 60 months currently permitted by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Option B, to a 75-month frequency for Type C leakage rate testing of selected components, in accordance with NEI 94-01, Revision 3-A (Reference 2).

- Adopt the use of ANSI/ANS 56.8-2002, "Containment System Leakage Testing Requirements." (Reference 43)

- Adopt a more conservative allowable test interval extension of nine months, for Type A, Type B and Type C leakage rate tests in accordance with NEI 94-01, Revision 3-A (Reference 2).

Specifically, the proposed change contained herein, would revise QCNPS TS 5.5.12 by replacing the reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," (Reference 1) with a reference to NEI 94-01, Revision 3-A (Reference 2), and the limitation and conditions specified in NEI 94-01, Revision 2-A, dated October 2008 (Reference 8). These new documents will be used by QCNPS to continue with the implementation of the performance-based leakage testing program in accordance with Option B of 10 CFR 50, Appendix J.

This License Amendment Request (LAR) also proposes an administrative change to TS 5.5.12 to delete the information regarding the performance of the next QCNPS Type A tests to be performed no later than July 22, 2009, for Unit 1 and May 16, 2008, for Unit 2, as these Type A tests have already occurred.

2.0 DETAILED DESCRIPTION

QCNPS TS 5.5.12, "Primary Containment Leakage Rate Testing Program," currently states, in part:

> This program shall establish the leakage 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 exemption. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995, as modified by the following exceptions:
>
> 1. NEI 94-01 – 1995, Section 9.2.3: The first Unit 1 Type A test performed after the July 23, 1994, Type A test shall be performed no later than July 22, 2009.
>
> 2. NEI 94-01 – 1995, Section 9.2.3: The first Unit 2 Type A test performed after the May 17, 1993, Type A test shall be performed no later than May 16, 2008.

The proposed changes to QCNPS TS 5.5.12 will replace the reference to RG 1.163 with a reference to NEI TR NEI 94-01 Revisions 2-A and 3-A.

Additionally, this LAR incorporates an administrative change to TS 5.5.12 to delete the information regarding the performance of the next QCNPS Type A tests to be performed no later than July 22, 2009, for Unit 1 and May 16, 2008, for Unit 2. This change will have no impact as these dates have already occurred and these Type A tests have already been performed. This Type A test information had been previously approved in Amendments No. 220 and No. 214 for QCNPS, Units 1 and 2, respectively, and is no longer applicable since the test dates occur in the past.

The proposed change will revise TS 5.5.12 to state, in part:

> This program shall establish the leakage 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 NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

A markup of the proposed change is provided in Attachment 2.

A plant specific risk assessment conducted to support this proposed change, summarized in Section 3.4 of this enclosure, is presented in full in Attachment 3 of this LAR. This risk assessment follows the guidelines of NRC RG 1.174, Revision 2 (Reference 3) and NRC RG 1.200, Revision 2 (Reference 4). The risk assessment concluded that increasing the ILRT test frequency 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 QCNPS risk profiles.

3.0 TECHNICAL EVALUATION

3.1 Description of Primary Containment System

QCNPS 1 and 2 were built with the General Electric Mark I primary containment system that is designed to condense the steam released during a postulated loss-of-coolant accident (LOCA), to limit the release of fission products associated with such an accident, and to serve as a source of water for the emergency core cooling system (ECCS).

Primary containment consists of a drywell, which encloses the reactor vessel, reactor coolant recirculation system, and branch lines of the reactor coolant system; a toroidal-shaped pressure suppression chamber containing a large volume of water (i.e., torus or wetwell); and a vent system connecting the drywell to the water space of the suppression chamber.

The safety design basis for the primary containment is to withstand the pressures and temperatures of the limiting design bases accident (DBA) without exceeding the design leakage rate. Primary containment is designed for a maximum internal pressure of 56 psig and for a maximum allowable internal operating pressure of 62 psig, both coincident with a maximum temperature of 281°F. The maximum allowable leakage rate for primary containment is $\leq$ 1.0 L_a, where L_a is defined as 3 percent of primary containment air weight per day at the design basis LOCA maximum peak containment pressure (P_a) of 43.9 psig.

The drywell is a steel pressure vessel with a spherical lower section, approximately 66 ft in diameter, a cylindrical upper section, approximately 37 ft in diameter and a hemispherical tophead. The drywell shell is enclosed in reinforced concrete to provide radiological shielding and additional resistance to deformation. A portion of the lower spherical drywell section is embedded in concrete. Beneath the drywell is a 26 ft thick concrete fill from the spring line down. Above the foundation transition zone, the drywell is separated from the reinforced concrete by a gap of approximately 2 inches to accommodate thermal expansion. The embedment, in combination with the upper lateral restraints attached to the cylindrical section, forms the drywell support system.

The suppression chamber is a steel pressure vessel, approximately 109 ft in diameter, constructed from 16 mitered cylindrical shell segments 30 ft in diameter, joined together to shape a torus, encircling and located below the drywell. It contains approximately 115,000 cubic feet of water and has a free air volume above the water line. The vertical support system provides a load transfer mechanism which acts to reduce local suppression chamber shell stresses and to more evenly distribute reaction loads to the reactor building basemat.

The drywell and suppression chamber are interconnected by a vent system. Eight main vents connect the drywell to a vent ring header, which is located within the suppression chamber air space. A bellows assembly is located at the junction where each main vent penetrates the suppression chamber shell to permit differential movement of the suppression chamber and drywell/vent system. Projecting downward from the vent ring header are downcomer pipes, arranged in 48 pairs around the vent header circumference, terminating below the surface of the suppression chamber water volume.

The original design of the Mark I containment system considered postulated accident loads associated with the containment design. These included pressure and temperature loads resulting from a LOCA, seismic loads, dead loads, jet-impingement loads, hydrostatic loads due to water in the suppression chamber, and pressure test loads. Subsequently, while performing large-scale testing for the Mark III containment system and in-plant testing for the Mark I primary containment system, new suppression chamber hydrodynamic loads were identified. Because these hydrodynamic loads had not been considered in the original design of the containment, a detailed re-evaluation was undertaken. This re-evaluation, referred to as the Mark I Program, involved tasks performed to restore the originally intended design safety margins for the QCNPS containment. The Mark I Program culminated in the issuance of the plant unique analysis report (PUAR) (Reference 21) for QCNPS followed by NRC review and acceptance (Reference 22).

Primary containment, including the suppression chamber for QCNPS, Units 1 and 2, were originally designed, erected, pressure-tested, and N-stamped in accordance with the ASME Code, Section III, 1965 Edition with Addenda up to and including Winter 1965.

For the Mark I Program re-evaluation, the acceptance criteria generally follow the rules contained in the ASME Code, Section III, 1977 Edition with Addenda up to and including Summer 1977 for Class MC (Metal Containment) components and component supports. Further detail regarding structural acceptance criteria may be found in the QCNPS Updated Final Safety Analysis Report (UFSAR) Section 3.8.2.3.5.

3.1.1 Pipe Penetrations

Two general types of pipe penetrations are provided in the QCNPS Mark I containment, they are: (1) those which must accommodate thermal movement, and; (2) those which experience relatively little thermal stress. The piping penetrations, which accommodate thermal movement, are the high temperature lines such as the steam lines, feedwater lines, and other reactor auxiliary system lines. The drywell nozzle passes through the concrete shield and is attached to a bellows expansion joint, which in turn, is attached to a penetration adapter to form a containment pressure boundary. The process line, which passes through the penetration, is attached to the penetration adapter and is free to move axially. A guard pipe immediately surrounds the process line and is designed to protect the bellows and containment boundary should the process pipe fail within the penetration.

Penetration details of piping lines that allow for relatively little movement are pipe sleeves that attach to the drywell. These penetrations are designed for 56 psig, but because of structural thicknesses, can withstand a substantially higher pressure. No bellows are required, since drywell thermal expansion is minimal.

3.1.2 Electrical Penetrations

Electrical penetrations were designed to accommodate the electrical requirements of the plant. Penetrations are functionally grouped into low voltage power and control cable penetration assemblies, high voltage power cable penetration assemblies, and shielded cable penetration assemblies. Each penetration seal has the same basic elements as shown in the QCNPS UFSAR Figure 3.8-39 (Reference 37).

An assembly is sized to be inserted in and welded to a 12-inch schedule 80 penetration nozzle, which were furnished as part of the containment structure. Installation of the penetration assembly was accomplished by inserting it from either side of the containment into the penetration nozzle. Three field welds were required to complete the installation of the assembly in the penetration nozzle.

The design and fabrication of each type of penetration assembly is in accordance with the requirements of the ASME Boiler and Pressure Code, Section III, Class B Vessel, and materials of construction are self-extinguishing in accordance with ASTM-D635.

3.1.3 Traversing In-Core Probe Penetrations

The traversing in-core probe (TIP) system, described in Section 7.6 of the UFSAR, has five guide tubes which pass from the reactor building through the primary containment. Guide tube penetrations of the primary containment are sealed by brazing which meets the requirements of ASME Boiler and Pressure Vessel Code, Section VIII.

3.1.4 Personnel and Equipment Access Locks

Access to the drywell is provided by the drywell head, one personnel airlock, one control rod drive removal hatch, and one bolted equipment hatch.

The personnel airlock has a locking mechanism on each door that is designed so that a tight seal will be maintained under either internal or external pressure. The doors are mechanically interlocked so that a door may be operated only if its companion door is closed and locked.

The hatch covers are bolted in place and sealed with a double tongue-and-groove seal. The seals on the hatches can be tested for leakage.

3.1.5 Pressure Suppression Chamber

Access to the pressure suppression chamber from the reactor building is provided by two access ports consisting of manholes with double-gasketed bolted covers. These access ports are bolted closed when primary containment integrity is required. They are opened only when the primary coolant temperature is below 212°F and the pressure suppression system is not required to be operational. A test connection between the double gaskets on each cover permits checking gasket leak tightness without pressurizing the containment. A drainpipe with double isolation valves provides for suppression chamber cleaning and decontamination.

3.1.6 Access for Refueling Operations

The drywell head is removed during refueling operations. The head is held in place by bolts and is sealed with a double tongue-and-groove seal arrangement, which permits periodic checks for leak tightness without pressurizing the entire containment. The head is bolted closed when primary containment integrity is required.

3.1.7 Modifications to Primary Containment

Although not a modification to the primary containment, a modification to the primary containment vent piping is underway at QCNPS, Units 1 and 2. This modification installs a hardened containment vent system (HCVS) to comply with NRC Order EA-13-109. This NRC Order is the result of lessons learned from the Fukushima Dai-ichi event. Specifically, EA-13-109 requires that boiling water reactors (BWRs) with Mark I or Mark II containments ensure that in addition to pre-core damage venting capability, the HCVS also provides a reliable hardened venting capability from the wetwell and drywell under severe accident conditions, including those involving a breach of the reactor vessel by molten core debris. Upon installation, this modification will be tested and maintained in accordance with the Appendix J and Containment ISI Programs as applicable.

The portion of the new HCVS does not interface with the existing Augmented Primary Containment Vent System required by NRC Generic Letter (GL) 89-16 as described in the QCNPS UFSAR Section 6.2.1.2.4.5.2.

Primary containment is also not impacted since the tie-in for the HCVS will be to the vent line outboard of an existing primary containment isolation valve. The modification installs a new valve second in-line valve as an outboard CIV in the existing vent line. The new valve, once installed, will be tested and become a part of the Appendix J Type C Local Leak Rate Test (LLRT) Program.

3.2 Emergency Core Cooling System Net Positive Suction Head Analysis (Post-Extended Power Uprate (EPU))

The ECCS, Residual Heat Removal (RHR), and Core Spray pump net positive suction head (NPSH) requirements are addressed in Section 6.3.3.2.9.3 of the UFSAR. An evaluation was conducted to support NPSH pumping requirement for post-extended power uprate (EPU) (i.e., 2957 MWth) operation. The analysis evaluated both short term (i.e., first 600 seconds) and long term (i.e., after 600 seconds) post-accident pressure and temperature response of containment. The containment analyses determined minimum containment pressure present in the suppression chamber air space for these bounding cases and support the use of the following credited containment pressure values (Table 3.2-1 below) used in the RHR and Core Spray NPSH analyses.

Table 3.2-1 Credited Containment Pressure		
From (Seconds)	To (Seconds)	Credited Containment Pressure (psig)
0	290	8.0
290	5,000	4.8
5,000	44,500	6.7
44,500	52,500	6.0
52,500	60,500	5.5
60,500	75,000	4.7
75,000	95,000	3.8
95,000	115,000	3.0
115,000	155,000	2.3
155,000	Accident End	1.8

The analysis showed sufficient containment pressure is available during the first 290 seconds to provide adequate NPSH for the RHR and Core Spray pumps; however, pump cavitation may occur for a short time after 290 seconds until operators throttle the RHR and Core Spray systems to restore NPSH. While the pumps may cavitate during this time period, they will continue to provide sufficient flow to the vessel to ensure core flood up. Cavitation tests have been performed on the RHR pump, which is the same model as the Core Spray pump, and these tests demonstrated that the pumps can cavitate in the short-term without any damage to pump internals or any degradation in pump performance.

The values shown in Table 3.2-1 above, and in UFSAR Section 6.3.3.2.9.3, for credited containment pressure in the RHR and Core Spray NPSH analyses, were evaluated by the NRC and approved in the safety evaluation (SE) for Amendments 202 and 198 for Units 1 and 2, respectively (Reference 15).

3.3 Justification for the Technical Specifications 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 surveillances 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 of the systems and components penetrating primary containment. The limitation on containment leakage provides assurance that the containment would perform its design function following an accident up to and including the plant 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 CIV leakage rates. Types B and C tests identify the vast majority of potential containment leakage paths. Type A tests identify the overall (i.e., 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 1) was issued. The RG endorsed NEI 94-01, Revision 0, (Reference 5) with certain modifications and additions. Option B, in concert with RG 1.163 and NEI 94-01, Revision 0, allows licensees with a satisfactory ILRT performance history (i.e., two consecutive, successful Type A tests) to reduce the test frequency for the containment Type A ILRT test from three tests in 10 years to one test in 10 years. This relaxation was based on an NRC risk assessment contained in NUREG-1493, (Reference 6) and Electric Power Research Institute (EPRI) TR-104285 (Reference 7), 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 8), 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 SE 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 1). 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.

In 2012, NEI 94-01, Revision 3-A (Reference 2), 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. NEI 94-01 has been endorsed by RG 1.163 and NRC SEs of June 25, 2008 (Reference 9), and June 8, 2012 (Reference 10), as an acceptable methodology for complying with the provisions of Option B in 10 CFR 50, Appendix J. The regulatory positions stated in RG 1.163 as modified by References 9 and 10 are incorporated in this document. 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 tests where the results of each test are within a licensee's allowable administrative limits. Intervals may be increased from 30 months up to a maximum of 120 months for Type B tests, except for containment airlocks, and up to a maximum of 75 months for Type C tests. If a licensee considers extended test intervals of greater than 60 months for Type B or Type C tested components, the review should include the additional considerations of as-found tests, schedule and review as described in NEI 94-01, Revision 3-A, Section 11.3.2.

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 SE Section 3.1.1.2 which 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.

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

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

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

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

3.3.2 Current QCNPS ILRT 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 January 11, 1996, the NRC approved amendments 169 and 165 for QCNPS Units 1 and 2, respectively, authorizing the implementation of 10 CFR 50, Appendix J, Option B for Types A, B and C tests (Reference 13).

In the implementation of Option B, the SE noted that QCNPS differed with the model TS developed by the NRC in cooperation with NEI, on one item. QCNPS chose to retain its existing surveillance to monitor secondary containment integrity. The NRC noted that: "The current specifications provide adequate assurance of secondary containment, were previously approved by the staff, and are acceptable. Based on the above, the licensee's proposed changes implementing Option B of Appendix J are acceptable." (Reference 13)

Option B states that specific existing exemptions to Option A are still applicable unless specifically revoked by the NRC. QCNPS currently has approved exemptions to 10 CFR 50, Appendix J that were issued by the NRC on June 12, 1984 (Reference 41). These exemptions, which focus on testing methodology aspects of Appendix J, are unaffected by the change to the Option B testing frequency requirements. These exemptions are also unaffected by the proposed change to the ILRT testing frequency.

Currently, TS 5.5.12 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, as modified by approved exemption. The program is required to be in accordance with the guidelines contained in RG 1.163. RG 1.163 endorses, with certain exceptions, NEI 94-01, Revision 0, 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 (Reference 5) rather than using test intervals specified in ANSI/ANS 56.8-1994. 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 consecutive periodic Type A tests where the calculated performance leakage was less than 1.0 L_a. 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" 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. The evaluation documented in NUREG-1493 included a study of the dependence or reactor accident risks on containment leak tightness for differing containment types. NUREG-1493 concluded in Section 10.1.2 that reducing the frequency of Type A tests from the original three tests per 10 years to one test per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Types B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements. Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, NUREG-1493 concluded that increasing the interval between ILRTs is possible with minimal impact on public risk.

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

<u>SE dated January 11, 1996 (ML021160123)</u>

The NRC approved amendments 169 and 165 for QCNPS Units 1 and 2, respectively, on January 11, 1996 (Reference 13). The amendment authorized the implementation of 10 CFR 50, Appendix J, Option B for Types A, B and C tests.

<u>SE dated December 21, 1999 (ML993630259)</u>

The NRC issued amendments 192 and 188 for QCNPS Units 1 and 2, respectively, on December 21, 1999 (Reference 14). The amendments changed TS 3/4.7.D and the associated Bases to eliminate the individual leakage limits for each main steam isolation valve (MSIV). The removed limits were replaced with a total limit for all four main steam lines combined. The current leakage limit is 11.5 standard cubic feet per hour (scfh) per valve. The amendments changed the limit to 46 scfh for all four main steam lines combined. The value chosen for the new total limit is equivalent to the sum of the current individual limits.

<u>SE dated December 21, 2001 (ML013540222)</u>

The NRC issued amendments 202 and 198 for QCNPS Units 1 and 2, respectively, on December 21, 2001 (Reference 15). The amendments allowed an increase in the maximum authorized operating power level from original rated thermal power (ORTP) of 2511 MWth to 2957 MWth. The changes increased the rated thermal power (RTP) by approximately 17.8 percent and were considered an EPU. The amendments changed the TS appended to the operating licenses to allow plant operation at 2957 MWth. These amendments also modified license conditions and requested additional license conditions to support the power uprate. Two noteworthy license changes of the EPU amendments with consideration to containment are: (1) decreasing P_a, from the pre-EPU peak calculated primary containment internal pressure from a DBA resulting in a P_a of 48.0 psig to post-EPU DBA P_a of 43.9 psig (SE, Section 4.1.1.3); and (2) containment overpressure is credited for pressure effects on NPSH for the RHR and Core Spray pumps (SE, Section 4.2.5) (See also Section 3.2 of this LAR).

<u>SE dated October 10, 2003 (ML032740364)</u>

The NRC issued amendments 218 and 212 for QCNPS Units 1 and 2, respectively, on October 10, 2003 (Reference 16). The amendments allowed a revision to TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," Surveillance Requirement (SR) 3.6.1.3.8 to require that a "representative sample" of reactor instrumentation line excess flow check valves (EFCVs) be tested every 24 months, such that each EFCV will be tested nominally at least once every

10 years. The Frequency of SR 3.6.1.3.8 is in accordance with the Surveillance Frequency Control Program, which currently requires performance of SR 3.6.1.3.8 on a 24-month frequency.

<u>SE dated March 8, 2004 (ML040280368)</u>

The NRC issued amendments 220 and 214 for QCNPS Units 1 and 2, respectively, on March 8, 2004 (Reference 17). These amendments provided a one-time TS change to extend the test interval from 10 to 15 years for the containment leakage rate Appendix J Type A tests. Additionally, these amendments included the following exceptions:

1. NEI 94-01-1995, Section 9.2.3: The first Unit 1 Type A test performed after the July 23, 1994, Type A test shall be performed no later than July 22, 2009; and

2. NEI 94-01-1995, Section 9.2.3: The first Unit 2 Type A test performed after the May 17, 1993, Type A test shall be performed no later than May 16, 2008.

Note: The LLRTs (Type B and Type C tests), including their schedules, were not affected by these amendments. In addition, the vacuum breaker TS SRs 3.6.1.7 and 3.6.1.8, including their schedules, were not affected by these amendments.

<u>Safety Evaluation Report (SER) dated October 28, 2004 (ML042960560)</u>

The NRC issued SER (NUREG-1796) related to the License Renewal of QCNPS, Units 1 and 2 on October 28, 2004 (Reference 18). This renewed license approves extended operation for both units until December 13, 2032. Per the SER, Section 2.4, Scoping and Screening Results: Structures, and found in Section 2.4.1.3, Conclusions, the NRC concluded that; …"the applicant has adequately identified the structural components of the primary containment that are within the scope of license renewal, as required by 10 CFR 54(a), and that the applicant adequately identified the structural components of the primary containment that are subject to AMR, as required by 10 CFR 54.21(a)(1)."

Additionally from UFSAR Appendix A, Section A.3.4, Containment Fatigue, it is noted that fatigue management activities will ensure that fatigue effects are adequately managed and are maintained within code design limits for extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii). UFSAR Section A.1.28 also credits the existing 10 CFR 50, Appendix J Program for monitoring leakage rates through the containment pressure boundary during the period of extended operation.

<u>SE dated September 11, 2006 (ML062070290)</u>

The NRC issued amendments 233 and 229 for QCNPS Units 1 and 2, respectively, on September 11, 2006 (Reference 40). These amendments approved adoption of an alternative source term methodology by replacing the current accident source term described in Technical Information Document (TID) 14844 (source term) with an accident source term as prescribed in 10 CFR 50.67, "Accident source term." Applicable parts of these amendments pertaining to containment and this LAR are: (1) a change to the maximum allowable containment leak rate from 1 percent primary containment air weight per day to 3 percent primary containment air weight per day (Section 3.3.6 of the SE), and (2) a change to the allowable leak rate limits for the MSIVs from 11.5 scfh individual/46 scfh combined to a new limit of 34 scfh individual/86 scfh combined (Section 3.3.7 of the SE).

3.3.4 QCNPS ILRT History

As noted previously, the QCNPS TS 5.5.12 currently requires Types A, B, and C testing in accordance with RG 1.163, which endorses the methodology for complying with 10 CFR 50, Appendix J, Option B. Since the adoption of Option B, the performance leakage rates are calculated in accordance with NEI 94-01, Section 9.1.1 for Type A testing. Table 3.3-1 lists the past periodic Type A ILRT results for QCNPS, Units 1 and 2.

Table 3.3-1
QCNPS Units 1 and 2 Type A ILRT Test History

Unit	Test Date	[1]Leakage 95% Upper Confidence Limit (wt%/Day)	[2]Total Leakage As Found (wt%/Day)	[3]Total Leakage As Left (wt%/Day)	[4]Acceptance Limit As Found/As Left (L_a)
1	March 22-23, 1986	0.2975	Note 9	Note 9	1.0/0.75
1 1	September 14, 1987[5] December 5-6, 1987	≈ 2.13 0.3508	- 1.0591[10]	- 0.4745	1.0/0.75 1.0/0.75
1	November 14-15, 1989	0.4480	5.412[11]	0.5411	1.0/0.75
1	Feb 28-Mar 2, 1991	0.6069	0.9185[12]	0.6853	1.0/0.75
1	December 5-8, 1992	0.2944	0.6926[13]	0.4015	1.0/0.75
1	July 23-24, 1994	0.3382	0.6168[14]	0.4082	1.0/0.75
1	May 18-19, 2009	0.6462	1.1419	0.9801	3.0/2.25[6]
2	May 26-28, 1985	0.4092	1.032[15]	0.549	1.0/0.75
2	October 12-13, 1986[7] October 14-15, 1986	0.9480 0.3618	1.2351 0.7614	- 0.4743	1.0/0.75 1.0/0.75
2	June 12-13, 1988	0.4621	3.497[16]	0.5409	1.0/0.75
2	April 27-28, 1990	0.4452	Notes 8,9 and 17	0.5330	1.0/0.75
2	April 1-6, 1992	0.2458	0.5555	0.3412	1.0/0.75
2	May 17-19, 1993	0.5064	0.7359[18]	0.6269	1.0/0.75
2	March 23, 2008	0.387	0.5992	0.5632	3.0/2.25[6]

Note 1: ILRT As Found test leakage value determined by end of test unadjusted 95% upper confidence limit. No Type B and C penalties or level change penalties were assigned or included in this leak rate test value.

Note 2: ILRT As Found leak rate test data contains 95% upper confidence limit and all penalties assigned including leakage value adjustments from Type B and C component repairs performed prior to ILRT test and level changes during ILRT test.

Note 3: ILRT As Left leak rate test data containing 95% upper confidence limit and all adjusted penalties (e.g., level changes and isolated volumes).

Note 4: The maximum primary containment leakage rate was ≤ 1.0 L_a for As Found ILRT testing and ≤ 0.75 L_a for As Left testing for startup. L_a was initially 1 percent of primary containment air weight per day, and was later revised to 3 percent of primary containment air weight per day (See Section 3.3.3 of this LAR for further description). Current TS leakage rate acceptance criteria as discussed in TS 5.5.12 for a Type A test for unit startup is 0.75 L_a (i.e., 2.25 percent containment air weight per day).

Note 5: ILRT performed at the beginning of September 1987 Unit 1 outage resulted in As Found ILRT failure. (LER 87-019 written). Follow-up ILRT performed at the end of the refueling passed leakage criteria.

Note 6: Allowable leakage criteria was changed from 1.0 L_a to 3.0 L_a (See Section 3.3.3 of this LAR SE dated September 11, 2006 for further description)

Note 7: ILRT performed on October 11-13, 1986 Unit 2 outage resulted in As Found ILRT failure. (LER 86-015 written). Drywell head gasket replaced and ILRT retested on October 14-15, 1986, and passed leakage criteria.

Note 8: As Found value was excessive Type C LLRT leakage. Some components when LLRT tested had an As Found leakage value beyond test equipment capability. Since As Found leakage was excessive, no comparative information can be provided for Noted item in Table.

Note 9: ILRT Test Data package does not provide this value or sufficient data to determine comparative information for the Table when considering As found and/or As left ILRT test results.

Note 10: Unit 1 LER 1987-016 written on excessive LLRT Leakage.

Note 11: Unit 1 LER 1989-014 written on excessive LLRT Leakage.

Note 12: Unit 1 LER 1990-029 written on excessive LLRT Leakage.

Note 13: Unit 1 LER 1992-020 written on excessive LLRT Leakage.

Note 14: Unit 1 LER 1994-005 written on excessive LLRT Leakage.

Note 15: Unit 2 LERs 1985-006 and 1985-007 written on excessive LLRT Leakage.

Note 16: Unit 2 LER 1988-007 written on excessive LLRT Leakage.

Note 17: Unit 2 LER 1990-003 written on excessive LLRT Leakage.

Note 18: Unit 2 LER 1993-007 written on excessive LLRT Leakage.

<u>Summary of ILRT Data History</u>

Table 3.3-1 presents the history of ILRT testing performed on each unit for approximately the past 30 years of operation at QCNPS. The third column of the Table identified as Leakage 95% Upper Confidence Limit (UCL) is test data without adjustments from any required penalties. This data indicates that the containment vessel performance, outside of the ILRT As Found failures in Unit 1 in 1987 and Unit 2 in 1986, has passed containment leakage acceptance criteria. The Unit 1 ILRT As Found cause of failure was not conclusively identified (LER 87-019). The Unit 2 ILRT As Found cause of failure determined a major source of leakage was a leaking drywell head gasket seal (LER 86-015). Shown in the fourth column of the Table, identified as Total Leakage As Found, are adjustments made to the UCL leakage based on Type B and C LLRT testing which resulted in a significant increase in the ILRT As Found leakage values. This increase is due to the poor performance of Type B and C tested components resulting in penalties adjusted against the As Found ILRT test results (LERs Noted). In the early 1990s, QCNPS took aggressive maintenance action to successively bring the sealing performance of LLRT test program components under control resulting in significantly lower As Found leakage from the troublesome components.

3.3.5 Drywell Bypass Leakage Rate Test

The leak tightness of the drywell is periodically verified by performance of the Drywell Bypass Leakage Rate Test (DBLRT). This test assists in the ongoing activities to monitor primary containment integrity by ensuring that the measured drywell bypass leakage is bounded by the safety analysis assumptions. The drywell integrity is further verified by a number of additional tests, including drywell airlock door seal leakage tests, overall drywell airlock leakage tests, drywell isolation valve tests and periodic visual inspections of exposed accessible interior and exterior drywell surfaces.

The DBLRT surveillance frequency and scheduling of TS SR 3.6.1.1.2 is controlled under the Surveillance Frequency Control Program (SFCP). As defined in QCNPS TS 5.5.14, "Surveillance Frequency Control Program," changes to the DBLRT frequency listed in the SCFP shall be made in accordance with NEI 04-10, "Risk-Informed Method for control of Surveillance Frequencies," Revision 1.

3.4 Plant Specific Confirmatory Analysis

3.4.1 Methodology

An evaluation has been performed to provide an assessment of the risk associated with implementing a permanent extension of the QCNPS containment Type A ILRT interval from ten years to fifteen years. The risk assessment follows the guidelines from a number of documents, which include: (1) NEI 94-01 (Reference 2), (2) the methodology outlined in EPRI TR-104285 (Reference 7) as updated by the EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (EPRI TR-1018243) (Reference 11), (3) the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in RG 1.174 (Reference 3), and (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 32). The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the EPRI TR-1018243 (Reference 11).

Details of the QCNPS Units 1 and 2 risk assessment, providing an assessment of the risk associated with implementing a permanent extension of the QCNPS containment Type A ILRT interval from ten years to fifteen years, are contained in Attachment 3.

The NRC report on performance-based leak testing, NUREG-1493 (Reference 6), analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a comparable BWR plant, that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Because ILRTs represent substantial resource expenditures, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures to support a reduction in the test frequency for QCNPS. The current analysis is being performed to confirm these conclusions based on QCNPS specific PRA models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 (Reference 7) methodology to perform the risk assessment. In October 2008, EPRI 1018243 (Reference 11) was issued to develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily NRC RG 1.174 (Reference 3). This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and containment conditional failure probability (CCFP), whereas EPRI TR-104285 considered only the change in risk based on the change in population dose. This ILRT interval extension risk assessment for QCNPS Unit 1 and Unit 2 employs the EPRI 1018243 methodology, with the affected system, structure, or component (SSC) being the primary containment boundary.

In the SE issued by NRC letter dated June 25, 2008 (Reference 9), the NRC concluded that the methodology in EPRI TR-1009325, Revision 2, was acceptable for referencing by licensees proposing to amend their TS to permanently extend the ILRT surveillance interval to 15 years, subject to the limitations and conditions noted in Section 4.0 of the SE. Table 3.4-1 below addresses each of the four limitations and conditions from Section 4.2 of the SE for the use of EPRI 1009325, Revision 2.

Table 3.4-1
EPRI Report No. 1009325 Revision 2 Limitations and Conditions

Limitation and Condition (From Section 4.2 of SE)	QCNPS 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 application.	QCNPS PRA technical adequacy is addressed in Section 3.4.2 of this LAR and Attachment 3, "Risk Assessment for QCNPS Regarding the ILRT (Type A) Permanent Extension Request," Appendix A, "PRA Technical Adequacy."

Table 3.4-1
EPRI Report No. 1009325 Revision 2 Limitations and Conditions

Limitation and Condition (From Section 4.2 of SE)	QCNPS Response
2.a The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small, and consistent with the clarification provided in Section 3.2.4.5 of this SE.	Since the ILRT extension has negligible impact on core damage frequency (CDF), the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT interval for the base case with corrosion included is 3.0E-08/yr, which falls within the "very small" change region of the acceptance guidelines in RG 1.174. If the EPRI Expert Elicitation methodology is used, the change is estimated as 6.56E-09/yr, which falls further within the very small change region of the acceptance guidelines in RG 1.174.
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% of the total population dose, whichever is less restrictive.	The change in dose risk for changing the Type A ILRT interval from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences for QCNPS, is 1.0E-02 person-rem/yr (0.31%) using the EPRI guidance with the base case corrosion included. The change in dose risk drops to 2.7E-03 person-rem/yr (0.08%) when using the EPRI Expert Elicitation methodology. The change in dose risk meets both of the related acceptance criteria for change in population dose of less than 1.0 person-rem/yr or less than 1% person-rem/yr.
2.c In addition, a small increase in 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 CCFP from the three in ten-year interval to one in fifteen years including corrosion effects using the EPRI guidance is 1.0%. This value drops to about 0.22% using the EPRI Expert Elicitation methodology. Both of these values are below the acceptance criteria of less than 1.5%.

Table 3.4-1
EPRI Report No. 1009325 Revision 2 Limitations and Conditions

Limitation and Condition (From Section 4.2 of SE)	QCNPS Response
3. The methodology in EPRI Report No. 1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate accident case (accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.	The representative containment leakage for Class 3b sequences used by QCNPS is 100 L_a, based on the recommendations in the latest EPRI report (Reference 20) and as recommended in the NRC SE on this topic (Reference 9). It should be noted that this is more conservative than the earlier previous industry ILRT extension requests, which utilized 35 L_a 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.	QCNPS relies upon containment over-pressure for ECCS performance. See Section 3.2 of this LAR Attachment for details. The RHR and Core Spray NPSH analyses were evaluated by the NRC and approved in the SE for Amendments 202 and 198 for Units 1 and 2, respectively.

3.4.2 Technical Adequacy of the PRA

The PRA Technical Adequacy evaluation is presented in Attachment 3, Appendix A, "PRA Technical Adequacy." The following is a summary of that evaluation.

3.4.2.1 Demonstrate the Technical Adequacy of the PRA

The guidance provided in RG 1.200 (Reference 4), Section 4.2, "License Submittal Documentation," indicates that the following items be addressed in documentation submitted to the NRC to demonstrate the technical adequacy of the PRA:

- Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.

- Document peer review findings and observations (F&Os) that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.

- Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the RG. Provide justification to show that where specific requirements in the standard are not met, it will not unduly impact the results.

- Identify key assumptions and approximations relevant to the results used in the decision-making process.

The risk assessment performed for the ILRT extension request is based on the current Level 1 and Level 2 PRA model. Note that for this application, the accepted methodology involves a bounding approach to estimate the change in the PRA risk metric of LERF from extending the ILRT interval. Rather than exercising the PRA model itself, it involves the establishment of separate evaluations that are linearly related to the plant CDF contribution. Consequently, a reasonable representation of the plant CDF that does not result in a LERF does not require that Capability Category II be met in every aspect of the modeling if the Category I treatment is conservative or otherwise does not significantly impact the results.

3.4.2.2 PRA Model Evolution and Peer Review Summary

The 2014A version of the QCNPS PRA model is the most recent evaluation of the Unit 1 and Unit 2 risk profile at QCNPS for internal event challenges. The QCNPS 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 QCNPS 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 EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the QCNPS PRA.

3.4.2.3 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 plant. This process is defined in the EGC Risk Management program, which consists of a governing procedure and subordinate implementation procedures. The PRA model update procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating EGC nuclear generation sites. The overall EGC Risk Management program defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.), and for controlling the model and associated computer files.

3.4.2.4 Plant Changes Not Yet Incorporated into the PRA Model

A PRA updating requirements evaluation (URE- EGC PRA model update tracking database) is created for all issues that are identified that could impact the PRA model. The URE database includes the identification of those plant changes that could impact the PRA model. A review of the open UREs indicates that there are no plant changes that have not yet been incorporated into the PRA model that would affect this application. FLEX modifications are in progress and will be incorporated in the QCNPS PRA in the future. The FLEX strategy will reduce CDF and is expected to lead to a reduction in the risk associated with the proposed ILRT extension. At this time, there is insufficient information to quantify the impact to this application, but the omission of FLEX credit in the model should in general result in added conservatism to the ILRT results.

3.4.2.5 Consistency with Applicable PRA Standards

Several assessments of technical capability have been made for the QCNPS internal events PRA models. These assessments are as follows and are further discussed in the paragraphs below.

- An independent PRA peer review (Reference 31) was conducted under the auspices of the BWR Owners' Group (BWROG) in 2000, following the Industry PRA Peer Review process (References 33 and 23). This peer review included an assessment of the PRA model maintenance and update process.

- In 2004, a gap analysis was performed to assess gaps between the peer review scope/detail of the Industry PRA Peer Review results relative to the available version of the ASME PRA Standard (Reference 29) and the draft version of RG 1.200, DG-1122 (Reference 4).

- During 2005 and 2006, the QCNPS PRA model results were evaluated in the BWROG PRA cross-comparisons study performed in support of implementation of the mitigating system performance index (MSPI) process (Reference 34).

- In January 2010, a self-assessment analysis was performed against the available version of the ASME/ANS PRA Standard (Reference 30) in preparation for the QCNPS 2010 PRA periodic update.

- In May 2010, an independent Focused PRA Peer Review (Reference 35) of the QCNPS Internal Flooding PRA model was performed using the NEI 05-04 process (Reference 46), the ASME/ANS PRA Standard (Reference 30), and RG 1.200, Rev. 2 (Reference 4).

- The QC 2010 self-assessment (Reference 36) was updated to incorporate the results of the final Focused PRA Peer Review report of the Internal Flooding PRA model.

- Following the most recent 2014 PRA update, another self-assessment (Reference 38) was performed to reflect the status after the 2014A model. This self-assessment was performed against the ASME/ANS PRA Standard (Reference 30), and RG 1.200, Rev. 2 (Reference 4).

- In February 2017, an independent PRA peer review (Reference 47) of the QCNPS Internal Events PRA model was performed using the NEI 05-04 Rev. 2 (Reference 46) process, the ASME PRA Standard (Reference 30), and RG 1.200, Rev. 2 (Reference 4). The peer review included all SRs except those related to internal flooding (which was previously peer reviewed in 2010). In addition, four SRs were assessed as not applicable to the QCNPS PRA. The results of that assessment are used as the basis for the capability assessment provided in Table A-2 of Attachment 3.

With the 2010 IF and 2017 peer reviews, all elements of the QCNPS PRA have undergone a thorough PRA peer review. The results of the most recent 2017 PRA peer review are as follows:

SR Capability

- 92% of the 259 applicable SRs are graded at Capability Category II or greater
- 3% of the SRs are graded at Capability Category I
- 5% of the SRs are graded as "Not Met"

Findings and Observations

There were 34 Findings. Table A-2 of Attachment 3, Section A.2.5 provides an assessment of each finding to the ILRT application.

3.4.2.6 Applicability of Peer Review Findings and Observations

Per the NRC SE for NEI 94-01, Revision 2 (Reference 9), the appropriate PRA quality to support an ILRT risk assessment is that the PRA Standard Supporting Requirements should meet Capability Category I or greater. There are 316 Technical Supporting Requirements plus 10 Maintenance and Update Supporting Requirements in the full power internal events (FPIE) portion of the ASME/ANS PRA Standard (Reference 30).

The 2010 Focused PRA Peer Review resulted in three findings, 10 suggestions and one best practice. Three supporting requirements were not met:

- IFSO-A3, IFSN-A7, and IFQU-A3

Table A-1 of Attachment 3 describes the findings associated with these SRs. The findings have been resolved and the findings have no impact to this application.

Per the 2017 QCNPS PRA peer review, there are 13 SRs that are not met:

- IE-C2, IE-C11, IE-C12, IE-D2, SY-A4, HR-G6, HR-G7, DA-C3, DA-C4, QU-B3, QU-C1, QU-E2, and QU-E4

The 2017 QCNPS PRA peer review identified seven SRs that are met at Capability Category I only:

- IE-B3, HR-D2, DA-D1, DA-D4, LE-C10, LE-C11, and LE-C12

The 2017 peer review findings are listed in Table A-2 of Attachment 3. The SRs associated with these findings are cross-referenced to the applicable findings within Table A-2. The 2017 peer review did not include a review of internal flooding SRs as this was performed in the 2010 Internal Flood focused peer review. The 2017 findings have not yet been resolved, but the potential impact upon the ILRT risk application results are assessed, as documented in Table A-2. No single finding was found to have a significant change to CDF if a model change was performed to address the finding. A number of finding resolutions will cause a small reduction in CDF. None of these findings are found to impact the conclusion of the ILRT risk application results. The cumulative impact of addressing all findings is judged to be minor and likely to reduce CDF.

3.4.2.7 External Events

Although EPRI report 1018243 (Reference 11) recommends a quantitative assessment of the contribution of external events (for example, fire and seismic) where a model of sufficient quality exists, it also recognizes that the external events assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval. Based on this, currently available information for external events models was referenced, and a multiplier was applied to the internal events results based on the available external events information. This is further discussed in Attachment 3, Risk Impact Assessment, Section 5.7, "External Events Contribution."

3.4.2.8 PRA Quality Summary

Based on the above, the QCNPS Units 1 and 2 PRA is of sufficient quality and scope for this application. The modeling is detailed; including a comprehensive set of initiating events (transients, LOCAs, and support system failures) including internal flood, system modeling, human reliability analysis and common cause evaluations. The QCNPS PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that these PRA models are suitable for use in the risk-informed process used for this application.

3.4.2.9 Identification of Key Assumptions

The methodology employed in this risk assessment followed the EPRI guidance (Reference 20) as previously approved by the NRC. The analysis included the incorporation of several sensitivity studies and factored in the potential impacts from external events in a bounding fashion. None of the sensitivity studies or bounding analyses indicated any source of uncertainty or modeling assumption that would have resulted in exceeding the acceptance guidelines. Since the accepted process utilizes a bounding analysis approach which is mostly driven by CDF contribution which does not already lead to LERF, there are no identified key assumptions or sources of uncertainty for this application (i.e., those which would change the conclusions from the risk assessment results presented here).

3.4.2.10 Summary

A PRA technical adequacy evaluation was performed consistent with the requirements of RG 1.200, Revision 2. This evaluation, combined with the details of the results of this analysis, demonstrates with reasonable assurance that the proposed extension to the ILRT interval for QCNPS Unit 1 and Unit 2 to fifteen years satisfies the risk acceptance guidelines in RG 1.174.

3.4.3 Summary of Plant-Specific Risk Assessment Results

The findings of the QCNPS, Unit 1 and 2 Risk Assessment contained in Attachment 3 confirm the general findings of previous studies that the risk impact associated with extending the ILRT interval from three in ten years to one in 15 years is small.

Based on the results from Attachment 3, Section 5.0, "Results," and the sensitivity calculations presented in Attachment 3, Section 6.0, "Sensitivities," the following conclusions regarding the assessment of the plant risk are associated with permanently extending the Type A ILRT test frequency to fifteen years:

- RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines "very small" changes in risk as resulting in increases of CDF below 1.0E-06/yr and increases in LERF below 1.0E-07/yr. "Small" changes in risk are defined as increases in CDF below 1.0E-05/yr and increases in LERF below 1.0E-06/yr. Since the ILRT extension was demonstrated to have negligible impact on CDF for QCNPS, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval for the base case with corrosion included is 3.0E-08/yr (Attachment 3 of this LAR, Table 5.6-1), which falls within the "very small" change region of the acceptance guidelines in RG 1.174.
    - When using the EPRI Expert Elicitation Methodology, the change is estimated as 6.6E-09/yr (Attachment 3 of this LAR, Table 6.2-2), which falls further within the very small change region of the acceptance guidelines in RG 1.174.
- The change in dose risk for changing the Type A test frequency from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences for QCNPS, is 1.0E-02 person-rem/yr (0.31%) using the EPRI guidance with the base case corrosion included (Attachment 3, Table 5.6-1). This change meets both of the related acceptance criteria for change in population dose of less than 1.0 person-rem/yr or less than 1% person-rem/yr identified in Attachment 3 of this LAR, Section 1.3.
    - When using the EPRI Expert Elicitation methodology, the change in dose risk drops to 2.7E-3 person-rem/yr (0.08%) (Attachment 3, Table 6.2-2). The change in dose risk meets both of the related acceptance criteria for change in population dose of less than 1.0 person-rem/yr or less than 1% person-rem/yr identified in Attachment 3 of this LAR, Section 1.3.
- The increase in the conditional containment failure frequency from the three in ten-year interval to one in fifteen years including corrosion effects using the EPRI guidance is 1.0% (Attachment 3, Section 5.5), which is below the acceptance criteria of 1.5% identified in Attachment 3 of this LAR, Section 1.3.
    - When using the EPRI Expert Elicitation methodology, this value drops to 0.22% (Attachment 3, Table 6.2-2). This value is below the acceptance criteria of less than 1.5% identified in Attachment 3 of this LAR, Section 1.3.
- To determine the potential impact from external events, a bounding assessment from the risk associated with external events was performed utilizing available information. As shown in Attachment 3, Table 5.7-6, the total increase in LERF due to internal events and the bounding external events assessment is 4.7E-07/yr. This value is in Region II of the RG 1.174 acceptance guidelines ("small" change in risk). The changes in dose risk

and conditional containment failure frequency also remained below the acceptance criteria.

- The same bounding analysis as shown in Attachment 3, Table 5.7-7, indicates that the total LERF from both internal and external risks is 4.2E-06/yr, which is less than the RG 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II ("small" change in risk).

- Including age-adjusted steel liner corrosion effects in the ILRT assessment was demonstrated to be a small contributor to the impact of extending the ILRT interval for QCNPS.

Therefore, 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 QCNPS risk profiles.

3.4.4 Previous Assessments

The NRC in NUREG-1493 (Reference 6) has previously concluded that:

- Reducing the frequency of Type A tests (i.e., ILRTs) from three per 10 years to one 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, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

The findings for QCNPS confirm these general findings on a plant specific basis for the ILRT interval extension considering the severe accidents evaluated for QCNPS, the QCNPS containment failure modes, and the local population surrounding QCNPS.

Details of the QCNPS, Units 1 and 2, risk assessment are contained in Attachment 3 of this LAR submittal.

3.5 Non-Risk Based Assessment

Consistent with the defense-in-depth philosophy discussed in RG 1.174, QCNPS has assessed other non-risk based considerations relevant to the proposed amendment. QCNPS has multiple inspections and testing programs that ensure the containment structure continues to remain 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 Coatings Inspection Program

QCNPS has committed to follow RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," Revision 0. The RG describes a method to comply with requirements of Appendix B to 10 CFR 50, and invokes several ANSI Standards. Standards pertinent to coatings are: ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," ANSI N101.4, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," and ANSI N5.12, "Protective Coatings for the Nuclear Industry."

QCNPS implements a safety-related coatings program that ensures DBA qualified coating systems are used inside Primary Containment. The program assures that safety-related DBA qualified coatings (Service Level 1) are selected, procured, applied and inspected in a manner that conforms to the applicable 10 CFR 50, Appendix B criteria. Unqualified coatings are controlled and tracked to ensure that emergency core cooling systems will not be adversely affected by coating debris following an accident. The program objective is to conform to licensee commitments made in response to GL 98-04. The safety-related coatings program also receives the support of the formal Maintenance Rule (10 CFR 50.65) condition-monitoring program. Engineering reviews and evaluates the results of coating condition examinations performed by qualified examiners.

A program to maintain containment coatings was developed to meet the requirements of RG 1.54, Revision 0 and is implemented by approved plant procedures.

Preventive maintenance activities have taken place and will continue to inspect and repair the protective coatings in the suppression chamber (submerged areas and vapor phase areas) and the drywell.

Primary Containment Interior Surface Coating Inspections are performed in the following areas: drywell liner interior surfaces, biological shield visible surfaces, subpile room surfaces, drywell head interior surface and the suppression chamber interior surface (including water line region and above).

The submerged regions of the suppression chamber have been routinely inspected in accordance with site and approved vendor procedures, and coating repairs have been proactively managed.

Table 3.5-1 provides results from coating inspections that have been performed on QCNPS Units 1 and 2, during the second CISI Interval for the past 3 refueling outages. The Table lists for each Unit's outage, the coating inspection results for both the Torus Underwater and the Primary Containment Interior Surface Coating inspections.

Table 3.5-1 QCNPS Units 1 and 2 Coating Inspection Results		
Refuel & Date	**Type of Inspection**	**Coating Inspection Results**
Unit 1		
Unit 1 Q1R21 May 2011	**Torus Underwater** Ref Document: 02-07-205.661, WO 01255482-02	100% of submerged shell inspected, desludged, 24 coating deficiencies found, no metal loss greater than 60 mil threshold (Reference 45), all repaired with Bio-Dur 561 Conclusion: Although small localized random failures (less than 1% of the total surface area) have occurred primarily due to fractured blisters and delaminations resulting in random spot corrosion and pitting, the balance of the coating is currently providing adequate protection of the substrate.
Unit 1 Q1R21 May 2011	**Coating Evaluation** Ref Document: Williams Specialty Services Report, May 9, 2011 WO 01255481-01, -02, -03, & -04.	Conclusion: Overall the Coating Systems throughout Unit 1 are showing wear relative to the age of the Plant. Generally speaking the Coating Systems are in GOOD condition. There are no imminent coating concerns that would negatively affect the safe shutdown or startup of the Plant. - Some areas repaired immediately and some are listed as recommendations for repairs. WO 01282915-01 repaired in Q1R21. WO's generated for resolving and tracking into Q1R22: 1478888, 1478890, 1478892, 1478893, 1478894, 1478883, 1468884, 1478886, 1478894 and 1478887.
Unit 1 Q1R22 March 2013	**Torus Underwater** Ref Document: 02-07-205.789, WO 01492232-02	100% of submerged shell inspected, desludged, 60 coating deficiencies found, no metal loss greater than 60 mil threshold, all repaired with Bio-Dur 561 Conclusion: Although random localized failures (less than 1% of the total surface area) have occurred, primarily due to fractured blisters and random spot pitting, the balance of the coating is currently providing adequate protection of the substrate.
Unit 1 Q1R22 March 2013	**Coating Evaluation** Ref Document: Williams Specialty Services Report, March 11, 2013	Conclusion: Overall, the Coating Systems throughout Unit 1 are in GOOD condition. Areas of concern have been identified in this and the previous Assessment Report. - Some areas were repaired immediately and some areas are listed as recommendations for repairs. Note: Q1R23 report discusses areas identified in previous coatings evaluation were repaired.
Unit 1 Q1R23 March 2015	**Torus Underwater** Ref Document: 02-14-233.75, WO 01649090-01 WO 01635232-02	100% of submerged shell inspected, desludged, 88 coating deficiencies found, no metal loss greater than 60 mil threshold, all repaired with Bio-Dur 561 – Conclusion: Although random localized small failures (less than 1% of the total surface area) have occurred, primarily due to random spot corrosion, the balance of the coating is currently providing adequate protection of the substrate and previously applied coating repairs are performing well.
Unit 1 Q1R23 March 2015	**Coating Evaluation** Ref Document: NUC2014435.04	Areas identified in previous coatings evaluation were repaired. Conclusion: The Q1R23 coating assessment identified areas of degraded coating requiring repair. No current coating conditions were identified that appear to affect structural integrity, plant operations, or the safe shutdown of the plant. - Some areas repaired immediately with remaining areas work orders written to address repair. (7 work orders for repair in Q1R24, they are: WO 01836118 thru WO 01836124)

Table 3.5-1 QCNPS Units 1 and 2 Coating Inspection Results		
Refuel & Date	**Type of Inspection**	**Coating Inspection Results**
Unit 2		
Unit 2 Q2R21 March 2012	**Torus Underwater** Ref Document: 02-07-205.722, WO 01333466-02	100% of submerged shell inspected, desludged, 1708 coating deficiencies found, no metal loss greater than 60 mil threshold, all repaired with Bio-Dur 561 Conclusion: Although small localized random failures (less than 1% of the total surface area) have occurred primarily due to zinc depletion of Carbo Zinc 11 SG spot repairs resulting in random spot corrosion and pitting, the balance of the coating is currently providing adequate protection of the substrate.
Unit 2 Q2R21 March 2012	**Coating Evaluation** Ref Document: Williams Specialty Services report, March 19, 2012 WO 01336189-01, 02, -03, & -04.	Conclusion: There are no immediate coating concerns that would impede the safe operation, start up or shut down of the plant. Some areas repaired immediately and some are listed as recommendations for repairs. WO's written for tracking into Q2R22: 1544425, 1538829-01, 1538829-02, 1538829-03, 1538829-04, 01549120-01, 01549120-02, 1574656, 1574637, 1574638, 1574655, 1574639, 1574654, 1574640, 1574641, 1574652, 1574653
Unit 2 Q2R22 April 2014	**Torus Underwater** Ref Document: 02-14-233.3, WO 01549120-01 & -022	100% of submerged shell inspected, desludged, 1675 coating deficiencies found, no metal loss greater than 60 mil threshold, all deficiencies repaired with Bio-Dur 561 Conclusion: Although small localized random failures (less than 1% of the total surface area) have occurred primarily due to zinc depletion of Carbo Zinc 11 SG spot repairs resulting in random spot corrosion and pitting, the balance of the coating is currently providing adequate protection of the substrate.
Unit 2 Q2R22 April 7, 2014	**Coating Evaluation** Ref Document: Williams Specialty Services report, April 7, 2014 WO 1538829-01, -02, -03, -04	Summary: There are no imminent coating concerns that would impede or prevent the safe operation, shutdown or startup of the Plant. WOs generated for resolving and tracking into Q2R23: 1751889-01, 1751890-01, 1751891-01, 1751893-01, 1751894-01, 1751895-01, 1751897-01, 1751896-01 and 175898-01
Unit 2 Q2R23 March 2016	**Torus Underwater** Ref Document: 02-14-233.145, WO 1750163-02	100% of submerged shell inspected, desludged, 1807 coating deficiencies found, no metal loss greater than 60-mil threshold, all deficiencies repaired with Bio-Dur 561 Conclusion: Although random localized small failures (less than 1% of the total surface area) have occurred primarily due to random spot corrosion, the balance of the coating is currently providing adequate protection of the substrate.

Table 3.5-1 QCNPS Units 1 and 2 Coating Inspection Results		
Refuel & Date	**Type of Inspection**	**Coating Inspection Results**
Unit 2 Q2R23 March 2016	**Coating Evaluation** Ref Document: NUC2016104 WO 01732678-01, -02, -03 & -04 Underwater Engineering Services, Inc. April 2016	All elevations of the Drywell Liner Plate, Drywell Head Interior, Torus Vapor Phase and Vent Header Interior were inspected to identify degraded coatings. Areas identified in previous coatings evaluations were repaired in accordance with UESI approved procedure and EGC specifications. Conclusion: The Q1R23 coating assessment identified areas of degraded coating requiring repair. No current coating conditions were identified that appear to affect structural integrity, plant operations, or the safe shutdown of the plant. There is considerable amount of mechanical damage to the liner plate coating due to scaffolding poles and insulation hitting the liner during outages. These areas are being continually repaired. WOs previously written for surface preparation and coating repair were 1751889-01, 1751890-01, 1751891-01, 1751893-01, 1751894-01, 1751895-01, 1751896-01, 1751897-01, 175898-01, 01751900-01, 01751901-01, 01751903-01, 01751904-01.

3.5.2 Containment Inservice Inspection Program

The Inservice Inspection (ISI) Program Plan details the requirements for the examination and testing of ISI Class 1, 2, 3, and MC pressure retaining components, supports, and containment structures at QCNPS, Units 1, 2, and Common (1/2). Unit Common components are included in the Unit 1 sections, reports, and tables. The ISI Program Plan also includes Containment Inservice Inspection (CISI), Risk-Informed Inservice Inspections (RISI), Augmented Inservice Inspections (AUG), and System Pressure Testing (SPT) requirements imposed on or committed to by QCNPS.

The ISI Program Plan is controlled and revised in accordance with the requirements of EGC procedure ER-AA-330, "Conduct of Inservice Inspection Activities," which implements the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI ISI Program.

The QCNPS, Units 1 and 2 are currently in the fifth ISI interval, which commenced on April 2, 2013, and ends on April 1, 2023. Additionally, QCNPS, Units 1 and 2 are in the second CISI interval, which started September 9, 2008, and is effective through September 8, 2018. These effective interval dates are based on QCNPS operating under an approved extended license renewal. The ASME Section XI code of record for the fifth ISI interval is the 2007 Edition through the 2008 Addenda, and the ASME Section XI code of record for the second CISI interval is the 2001 Edition through the 2003 Addenda.

The QCNPS CISI Plan includes ASME Section XI ISI Class MC pressure retaining components and their integral attachments that meet the criteria of Subarticle IWA-1300. This CISI Plan also includes information related to augmented examination areas, component accessibility, and examination review.

QCNPS has no ISI Class Concrete Containment (CC) components that meet the criteria of Subarticle IWL-1100; therefore, no requirements to perform examinations in accordance with Subsection IWL are incorporated into this CISI Plan.

The second interval CISI Program Plan was developed in accordance with 10 CFR 50.55a and the 2001 Edition through the 2003 Addenda of ASME Section XI, subject to the limitations and modifications contained within paragraph (b) of the regulation. These limitations and modifications are detailed in Table 3.5-2 of this section. Overall, this second interval CISI Program Plan addresses Subsections IWE, Mandatory Appendices of ASME Section XI, approved IWE Code Cases, approved alternatives through relief requests and SEs, and utilizes Inspection Program B (described in Section XI, IWE-2412).

Table 3.5-2
10 CFR 50.55a Requirements
(Applicable to Containment Inspection Program)

10 CFR 50.55a Paragraphs	Limitations, Modifications, and Clarifications
10 CFR 50.55a(b)(2)(ix)(A)	(CISI) *Examination of metal containments and the liners of concrete containments:* For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report as required by IWA-6000: *(1)* A description of the type and estimated extent of degradation, and the conditions that led to the degradation; (2) An evaluation of each area, and the result of the evaluation, and; *(3)* A description of necessary corrective actions.
10 CFR 50.55a(b)(2)(ix)(B)	(CISI) Examination of metal containments and the liners of concrete containments: When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased, provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.
10 CFR 50.55a(b)(2)(ix)(F)	(CISI) Examination of metal containments and the liners of concrete containments: VT-1 and VT-3 examinations must be conducted in accordance with IWA-2200. Personnel conducting examinations in accordance with the VT-1 or VT-3 examination method shall be qualified in accordance with IWA-2300. The "owner-defined" personnel qualification provisions in IWE-2330(a) for personnel that conduct VT-1 and VT-3 examinations are not approved for use.
10 CFR 50.55a(b)(2)(ix)(G)	(CISI) *Examination of metal containments and the liners of concrete containments:* The VT-3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table IWE-2500-1, and the VT-1 examination method must be used to conduct the examination in Item E4.11 of Table IWE-2500-1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE-2500-1 using the VT-3 examination method must be conducted once each interval. The "owner-defined" visual examination provisions in IWE-2310(a) are not approved for use for VT-1 and VT-3 examinations.
10 CFR 50.55a(b)(2)(ix)(H)	(CISI) *Examination of metal containments and the liners of concrete containments:* Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE-2500-1 must be examined using the VT-3 examination method. Flaws or degradation identified during the performance of a VT-3 examination must be examined in accordance with the VT-1 examination method. The criteria in the material specification or IWB-3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT-3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

Table 3.5-2 10 CFR 50.55a Requirements (Applicable to Containment Inspection Program)	
10 CFR 50.55a Paragraphs	**Limitations, Modifications, and Clarifications**
10 CFR 50.55a(b)(2)(ix)(I)	(CISI) *Examination of metal containments and the liners of concrete containments:* The ultrasonic examination acceptance standard specified in IWE-3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.

The inspection of containment structures and components are performed per procedures ER-AA-330-007, "Visual Examination of Section XI Class MC Surfaces and Class CC Liners," ER-AA-335-004, "Manual Ultrasonic Measurement of Material Thickness and Interfering Conditions," and ER-AA-335-018, "Visual Examination of ASME IWE Class MC and Metallic Liners of IWL Class CC Components."

Since both of the QCNPS units are in the second interval, CISI inspections have been completed for the first and second periods, with the third period inspections currently ongoing. The results of recent inspections performed during refueling outages, show that various indications were observed, documented, evaluated, and determined to be acceptable. The results of inspections performed during the past three refueling outages examining primary containment are summarized in Table 3.5-3 shown below.

There will be no change to the schedule for these inspections as a result of the extended ILRT interval.

Table 3.5-3 Summary of IWE Examinations on Primary Containment			
Unit 1			
Q1R21 IWE Inspections on WO 01244920-02; May 2011 – Per Exam schedule only inspections required were those for preservice and of repair/replacement activities.			
Database Component No.	Item	Results of Inspect	Resolution
None			
Q1R22 IWE Inspections on WO 01453925-02; March 2013 – Per Exam schedule only inspections required were those for preservice and of repair/replacement activities.			
Database Component No.	Item	Results of Inspect	Resolution
None			
Q1R23 IWE Inspections on WO 01633313-02; March 2015 – Inspected: Containment liner, Vent Header Interior – All bays & interior surfaces, Torus Interior (normally wetted area), X-2 Hatch area under moisture barrier after removal. Two recordable indications were noted for further action, as follows:			
Database Component No.	Item	Results of Inspect	Resolution
1X-2 PALF-Moisture Barrier	X-002 Floor Moisture Barrier	Recordable Indication – Moisture Barrier cracked and missing in spots	IR 2463927; replaced barrier and passed reinspection in WO 01812660-01
1-1600-T Torus	Bay 13	Recordable Indication – small area found with coating and primer missing inside vent header	Evaluated as acceptable. Wrote AR 02471483 to repair in Q1R24.

<table>
<tr><th colspan="4">Table 3.5-3
Summary of IWE Examinations on Primary Containment</th></tr>
<tr><th colspan="4">Unit 2</th></tr>
<tr><td colspan="4">Q2R21
IWE Inspections on WO 01337032-01 thru -06; March 2012 – (1) Exam conducted outside of Primary Containment. Area included Reactor Building, Basement, top of torus and refuel floor where the drywell head is located after removal; (2) Exam conducted inside of the torus vent header (centipede), from torus catwalk and of torus below the waterline; (3) Exam conducted from inside of the drywell. Applicable areas include all levels of the drywell. (Basement through 4th level). Results from all these inspections: no issues identified.</td></tr>
<tr><td>Database Component No.</td><td>Item</td><td>Results of Inspect</td><td>Resolution</td></tr>
<tr><td>None</td><td></td><td></td><td></td></tr>
<tr><td colspan="4">Q2R22
IWE Containment Inspections on WO 01549517-01, April 2014 – Exam conducted inside (all levels) and outside of both the drywell and the torus. Results: 17 Recordable Indications noted, 15 were characterized mostly as coating missing/surface corrosion with no metal loss and dispositioned with no action required. Two were noted for further action, they are:</td></tr>
<tr><td>Database Component No</td><td>Item</td><td>Results of Inspect</td><td>Resolution</td></tr>
<tr><td>2X-2 PALF-Moisture Barrier</td><td>X-002 Floor Moisture Barrier</td><td>Recordable Indication</td><td>IR1647016; replaced barrier and passed reinspection</td></tr>
<tr><td>2TORLB01 – ESURFACE</td><td>Torus Shell</td><td>Recordable Indication – Coating blister with water behind it, Bay 1</td><td>IR 1644607; removed blisters no metal loss, recoated</td></tr>
<tr><td colspan="4">Q2R23
IWE Containment Inspections on WO 01746044-02, March 2016 – Results: Exam conducted inside (all levels) and outside of both the drywell and the torus. Results: 16 Recordable Indications noted that were characterized as surface corrosion with no metal loss. All were dispositioned with no action required.</td></tr>
<tr><td>Database Component No</td><td>Item</td><td>Results of Inspect</td><td>Resolution</td></tr>
<tr><td>None</td><td></td><td></td><td></td></tr>
</table>

Programmatically, the 10-year CISI interval is divided into three successive inspection periods as determined by calendar year of plant service within the inspection interval. Table 3.5-4 identifies the period start and end dates for the second CISI interval as defined by Inspection ISI Program Plan. Table 3.5-5 identifies the successive period start and end dates for the third CISI interval, whose dates are approximate since the third CISI interval inspection program has not been developed at this time.

In accordance with paragraph IWA-2430(c)(1) of ASME Section XI, the inspection periods specified in these tables may be decreased or extended by as much as one year to coincide with refueling outages, and paragraph IWA-2420(d) allows an inspection interval to be extended when a unit is out of service continuously for six months or more. The extension may be taken for a period of time not to exceed the duration of the outage.

<table>
<tr><th colspan="7">Table 3.5-4
Units 1 and 2 Second CISI Interval/Period/Outage Matrix
(For ISI Class MC Component Examinations)</th></tr>
<tr><th colspan="2">Unit 1</th><th>Period</th><th>Interval</th><th>Period</th><th colspan="2">Unit 2</th></tr>
<tr><th>Outage Number</th><th>Outage or Projected Start Date</th><th>Start Date to End Date</th><th>Start Date to End Date</th><th>Start Date to End Date</th><th>Outage or Projected Start Date</th><th>Outage Number</th></tr>
<tr><td>Q1R20</td><td>April 2009</td><td rowspan="2">1^{ST}
9/9/08 to 9/8/11</td><td rowspan="5">2^{nd}
(Unit 1)
9/9/08 to 9/8/18

2^{nd}
(Unit 2)
9/9/08 to 9/8/18</td><td>1^{ST}
9/9/08 to 9/8/11</td><td>March 2010</td><td>Q2R20</td></tr>
<tr><td>Q1R21</td><td>May 2011</td><td rowspan="2">2^{nd}
9/9/11 to 9/8/15</td><td>March 2012</td><td>Q2R21</td></tr>
<tr><td>Q1R22</td><td>March 2013</td><td rowspan="2">2^{nd}
9/9/11 to 9/8/15</td><td>April 2014</td><td>Q2R22</td></tr>
<tr><td>Q1R23</td><td>March 2015</td><td rowspan="2">3^{rd}
9/9/15 to 9/8/18</td><td>March 2016</td><td>Q2R23</td></tr>
<tr><td>Q1R24</td><td>Scheduled 3/17</td><td>3^{rd}
9/9/15 to 9/8/18</td><td>Scheduled 3/18</td><td>Q2R24</td></tr>
</table>

<table>
<tr><th colspan="7">Table 3.5-5
Units 1 and 2 Third CISI Interval/Period/Outage Matrix
(For ISI Class MC Component Examinations)
(Approximate)[1]</th></tr>
<tr><th colspan="2">Unit 1</th><th>Period</th><th>Interval</th><th>Period</th><th colspan="2">Unit 2</th></tr>
<tr><th>Outage Number</th><th>Outage or Projected Start Date</th><th>Start Date to End Date</th><th>Start Date to End Date</th><th>Start Date to End Date</th><th>Outage or Projected Start Date</th><th>Outage Number</th></tr>
<tr><td>Q1R25</td><td>Scheduled 3/19</td><td rowspan="2">1^{ST}
9/9/18 to 9/8/21</td><td rowspan="3">2^{nd}
(Unit 1)
9/9/18 to 9/8/28</td><td>1^{ST}
9/9/18 to 9/8/21</td><td>Scheduled 3/20</td><td>Q2R25</td></tr>
<tr><td>Q1R26</td><td>Scheduled 3/21</td><td rowspan="2">2^{nd}
9/9/21 to 9/8/25</td><td>Scheduled 3/22</td><td>Q2R26</td></tr>
<tr><td>Q1R27</td><td>Scheduled 2/23</td><td>2^{nd}
9/9/21 to</td><td>Scheduled 3/24</td><td>Q2R27</td></tr>
</table>

Table 3.5-5 Units 1 and 2 Third CISI Interval/Period/Outage Matrix (For ISI Class MC Component Examinations) (Approximate)[1]						
Unit 1		**Period**	**Interval**	**Period**	**Unit 2**	
Outage Number	**Outage or Projected Start Date**	**Start Date to End Date**	**Start Date to End Date**	**Start Date to End Date**	**Outage or Projected Start Date**	**Outage Number**
Q1R28	Scheduled 2/25	9/8/25	2nd (Unit 2) 9/9/18 to 9/8/28	3rd 9/9/25 to 9/8/28	Scheduled 3/26	Q2R28
Q1R29	Scheduled 2/27	3rd 9/9/25 to 9/8/28			Scheduled 3/28	Q2R29

Note 1: Table 3.5-5 identifies the successive periods start and end dates for the Third CISI Interval, which is approximate since the Third CISI Interval inspection program has not been developed at this time

The QCNPS Containment ISI Plan includes ASME Section XI ISI Class MC pressure retaining components and their integral attachments that meet the criteria of Subarticle IWA-1300. This Containment ISI Plan also includes information related to examined areas, augmented examination areas, component accessibility, and examination review. A summary of inspected containment components, Category E-A and augmented containment components, Category E-C, are provided for Units 1 and 2 in Table 3.5-6.

Table 3.5-6 Units 1 and 2 IWE Inservice Inspection Summary						
Examination Category (with Examination Category Description)	**Item Number**	**Description**	**Exam Requirements**	**Total Number of Components (Unit 1 includes common)**	**Relief Request/ TAP Number**	**Notes**
E-A Containment Surfaces	E1.11	Containment Vessel Pressure Retaining Boundary – Accessible Surface Areas	General Visual	Unit 1: 295 Unit 2: 294		
	E1.11	Containment Vessel Pressure Retaining Boundary – Bolted Connections, Surfaces	Visual, VT-3	Unit 1: 69 Unit 2: 70		10 10
	E1.12	Containment Vessel Pressure Retaining Boundary – Wetted Surfaces of Submerged Areas	Visual, VT-3	Unit 1: 16 Unit 2: 16		11 11

Table 3.5-6 Units 1 and 2 IWE Inservice Inspection Summary						
Examination Category (with Examination Category Description)	**Item Number**	**Description**	**Exam Requirements**	**Total Number of Components** (Unit 1 includes common)	**Relief Request/ TAP Number**	**Notes**
	E1.20	Containment Vessel Pressure Retaining Boundary – BWR Vent System Accessible Surface Areas	Visual, VT-3	Unit 1: 121 Unit 2: 113		11 11
	E1.30	Containment Vessel Pressure Retaining Boundary – Moisture Barriers	General Visual	Unit 1: 4 Unit 2: 4		
E-C Containment	E4.11	Containment Surface Areas - Visible Surfaces	Visual, VT-1	Unit 1: 0 Unit 2: 3		12
Surfaces Requiring Augmented Examination	E4.12	Containment Surface Areas - Surface Area Grid	Ultrasonic	Unit 1: 0 Unit 2: 0		

Note 10: Bolted connections examined per Item Number E1.11 require a General Visual examination each period and a VT-3 visual examination once per interval and each time the connection is disassembled during a scheduled Item Number E1.11 examination. Additionally, a VT-1 visual examination shall be performed if degradation or flaws are identified during the VT-3 visual examination. These modifications are required by 10 CFR 50.55a(b)(2)(ix)(G) and 10 CFR 50.55a(b)(2)(ix)(H).

Note 11: Item Numbers E1.12 and E1.20 require VT-3 visual examination in lieu of General Visual examination, as modified by 10 CFR 50.55a(b)(2)(ix)(G).

Note 12: Item Number E4.11 requires VT-1 visual examination in lieu of Detailed Visual examination, as modified by 10 CFR 50.55a(b)(2)(ix)(G).

An additional monitoring of the containment liner applicable to QCNPS, was the inspections instituted at Dresden Nuclear Power Station Unit 3 of the inaccessible annulus area to ensure that potential corrosion does not occur. As part of Plant Licensing Renewal, NUREG-1796 (Reference 18), Section 3.0, Aging Management Review, page 3-403, a description is provided of the monitoring at Dresden consisting of the inspection of a sample of locations in the cylindrical and upper spherical areas of the drywell, using ultrasonic measurements of the drywell shell thickness made from accessible areas of the drywell interior. QCNPS Units 1 and 2 as well as Dresden Unit 2 credit the inspections performed on Dresden Unit 3 to establish the most conservative bounding case for continued inspection. This inspection is a part of the ASME Section XI, Subsection IWE Program, commitment B.1.26 at QCNPS.

3.5.2.1 Code Cases

The only Code Case implemented in the QCNPS containment ISI Program is N-649, which is an EGC fleet relief request identified as I5R-13. This relief request is briefly shown on Table 3.5-7 and is further described in detail below Table 3.5-7.

3.5.2.2 Relief Requests

Table 3.5-7 contains an index of Relief Requests applicable to the CISI Program. Note that only Relief Requests applicable to the requirements for Class MC components are addressed in this Table.

Table 3.5-7
Second Ten-Year CISI Interval Relief Request

Relief Request	Revision/ Date	Status	(Program) Description/ Approval Summary
I5R-13 EGC Fleet Relief Request	0 September 5, 2014	Authorized	Examination to Utilize ASME Code Case N-649, Revision 0. Alternative Requirements for IWE-5240 Visual Examination. RG 1.147, Revision 16. Authorized April 30, 2014

Explanation of use of Code Case N-649 in Accordance with 10 CFR 50.55a(a)(3)(i):

ASME Section XI, paragraph IWE-5240, "Visual Examination," requires that a detailed visual examination (IWE-2310) be performed during an IWE-5220 required pressure test on areas affected by repair/replacement activities.

ASME Code Case N-649, "Alternative Requirements for IWE-5240 Visual Examination Section XI, Division 1," allows for a VT-3, VT-1, general visual or detailed visual examination depending on the timing of the pressure test.

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

ASME Section XI, paragraph IWE-5240 requires that a detailed visual examination of repaired areas be completed during a post repair pressure test performed subsequent to IWE repairs. ILRTs required by 10 CFR 50, Appendix J, are often performed following repairs in order to fulfill the post-repair testing requirement. However, the IWE-5240 visual examination cannot be performed because the containment liners/shell are inaccessible during the post repair pressure tests (i.e., personnel are not able to be inside the containment during the ILRT).

In recognition of the inability to perform visual examinations of containment liners/shells during the post repair pressure test required by paragraph IWE-5240, ASME Code Case N-649 was issued to allow the visual examination to be performed during or after the pressure test on the areas affected by the repair/replacement activity. ASME Section XI did not address this inability in the Code until the 2004 Edition through the 2006

Addenda was issued; therefore, ASME Code Case N-649 is needed when using the 2001 Edition through the 2003 Addenda of ASME Section XI, which is applicable to Quad Cities Nuclear Generating Station.

The "Applicability Index for Section XI Cases," states that ASME Code Case N-649 is applicable up to and including the 1998 Edition with the 2000 Addenda of ASME Section XI; however, the code of record for QCNPS CISI Program is the 2001 Edition through the 2003 Addenda thus necessitating the need for this relief request. The Edition/Addenda referenced in the Code Case text itself also ends at the 1998 Edition with the 2000 Addenda. However, the requirements of paragraph IWE-5240 are identical in both the 1998 Edition through the 2000 Addenda and the 2001 Edition through the 2003 Addenda. Therefore, it is concluded that the proposed alternative provides an acceptable level of quality and safety.

The EGC Fleet relief request I5R-13, inclusive of QCNPS, requested that the applicability of ASME Code Case N-649 be extended to the 2001 Edition through 2003 Addenda for use during the station's second CISI interval. This relief request was subsequently authorized with an SE issued by the NRC on April 30, 2014. Additionally, NRC RG 1.147, Revision 16, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," lists ASME Code Case N-649 as acceptable for use with no conditions or limitations. No technical changes are being made to the Code Case.

3.5.2.3 Identification of Class MC and/or CC Exempt Components

The containment section of the ISI Classification Basis Document discusses the containment design and components. Metal containment surface areas subject to accelerated degradation and aging require augmented examination per Examination Category E-C and paragraph IWE-1240.

The CISI components overall were evaluated for potential candidates to be included programmatically in the Augmented Inspection Program. The details of this evaluation are contained in the ISI Classification Basis Document, Section 4.1.12. The evaluation resulted in no components being recommended on a programmatic basis, for the Augmented Program within Examination Category E-C, that would appear on table IWE-2500-1.

A significant condition is a condition that is identified as requiring application of additional augmented examination requirements under paragraph IWE-1240.

In the First CISI Interval, during the QCNPS Unit 2 Outage, Q2R19 Torus underwater IWE examinations, recordable indications were identified on the surface areas in the Torus Shell at Bays 3, 6, and 16. Portions of the Torus surface area near these Bays have been identified as augmented surface areas requiring examination in accordance with paragraph IWE-1240. These surface areas have been categorized in accordance with Table IWE-2500-1, Examination Category E-C, Item Number E4.11, requiring visual examination of 100% of the surface areas identified during each inspection period until the areas examined remain essentially unchanged for the next three inspection periods. In the Second CISI Interval, augmented surface areas require visual examination of 100% of the surface areas identified during each inspection period until the areas examined remain essentially unchanged for the next inspection period. Once an augmented area remains unchanged for one full period, the areas fall back to the normal Examination Category E-A examination schedule.

The second CISI Interval coating examinations performed during the Units 1 and 2 previous 3 refueling outages, are discussed in the previous section of this report (Section 3.5.1) and are summarized in Table 3.5-1. The augmented inspection area is the wetted (i.e., immersion zone) and submerged portions of the suppression chamber. These areas have undergone examinations to quantify and evaluate coating problems and pitting. The inspections found coating deficiencies with no metal loss greater than the defined pre-established acceptance criteria(Reference 45). All deficiencies were repaired during each inspection before unit startup.

3.5.2.4 Augmented Inspection Program Requirements

Augmented Inspection Program requirements are those inspections that are performed above and beyond the requirements of ASME Section XI.

Below is a summary of those examinations performed by QCNPS that are not specifically addressed by ASME Section XI, or the inspections that will be performed in addition to the requirements of ASME Section XI on a routine basis during the Second CISI Interval.

Note that per NUREG-1796, (Reference 18) QCNPS will perform a VT-3 visual examination on nonexempt Class MC piping supports, which were added to the augmented inspection program in accordance with the QCNPS commitment for license renewal. These inspections are addressed in the 5th Interval ISI Program Plan, and in the ISI Program Selection documents. The inspections are identified as NUREG-1796 inspections found in the ISI Database that are addressed by the IWF Program and by the Structural Monitoring Program.

The Augmented Inspection Plans resulting from past inspections at QCNPS associated with IWE and the integrity of the primary containment, are listed in Table 3.5-8 below.

Table 3.5-8
Units 1 and 2 Augmented Containment Inspection Program Matrix

Examination Category (with Examination Category Description)	Aug Number	Description	Exam Requirements	Total Number of Components	Relief Request/ TAP Number	Notes
E-C Containment Surfaces Requiring Augmented Examination	E4.11	Containment Surface Areas – Torus Bays 3, 6, and 16	Visual, VT-1	Unit 1: 0 Unit 2: 3	N/A	12

Note 12: Item Number E4.11 requires VT-1 visual examination in lieu of Detailed Visual examination, as modified by 10 CFR 50.55a(b)(2)(ix)(G).

3.5.2.5 Component Accessibility

ISI Class MC components subject to examination shall remain accessible for either direct or remote visual examination from at least one side per the requirements of ASME Section XI, paragraph IWE-1230.

Paragraph IWE-1231(a)(3) requires 80% of the pressure-retaining boundary that was accessible after construction to remain accessible for either direct or remote visual examination, from at least one side of the vessel, for the life of the plant. QCNPS addressed, in Calculation QDC-1600-M-1617 (Reference 12), compliance with this requirement by calculating the containment pressure boundary surface area that was accessible for examination at the beginning of the CISI Program resulting in a determination for the limit of surface area which may be made inaccessible for the balance of plant life.

Portions of components embedded in concrete or otherwise made inaccessible during construction are exempted from examination, provided that the requirements of ASME Section XI, paragraph IWE-1232 have been fully satisfied.

In addition, inaccessible surface areas exempted from examination include those surface areas where visual access by line of sight with adequate lighting from permanent vantage points is obstructed by permanent plant structures, equipment, or components; provided these surface areas do not require examination in accordance with the inspection plan, or augmented examination in accordance with paragraph IWE-1240.

3.5.2.6 Inaccessible Areas

For Class MC applications, QCNPS shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, QCNPS shall provide the following in the Owners Activity Report-1, as required by 10 CFR 50.55a(b)(2)(ix)(A):

- A description of the type and estimated extent of degradation, and the conditions that led to the degradation;
- An evaluation of each area, and the result of the evaluation; and
- A description of necessary corrective actions.

An evaluation has been performed to determine if QCNPS has inaccessible areas that could indicate the presence of or result in, degradation to such inaccessible areas requiring identification per 10 CFR 50.55a(b)(2)(ix)(A). The evaluation resulted in no areas identified and is contained in the ISI Classification Basis Document, Section 4.1.12.

QCNPS has not needed to implement any new technologies to perform inspections of any inaccessible areas at this time. However, EGC actively participates in various nuclear utility owner's groups and ASME Code committees to maintain cognizance of ongoing developments within the nuclear industry. Industry operating experience is also continuously reviewed to determine its applicably to QCNPS. Adjustments to inspection plans and availability of new, commercially available technologies for the examination of the inaccessible areas of the containment would be explored and considered as part of these activities.

3.5.2.7 Responsible Individual

ASME Section XI, Subsection IWE requires the Responsible Individual to be involved in the development, performance, and review of the CISI examinations. At QCNPS, the Responsible Individual is committed to meet the requirements of ASME Section XI, paragraph IWE-2320.

3.5.2.8 Examination Methods & Personnel Qualifications

The examination methods used to perform Code examinations for the nonexempt Class MC components are in accordance with 10 CFR 50.55a requirements and the applicable ASME Codes.

Personnel performing IWE examinations shall be qualified in accordance with EGC's written practice, or approved vendor written practice for certification and qualification of nondestructive examination personnel.

3.5.3 Supplemental Inspection Requirements

With the implementation of the proposed change, TS 5.5.12 will be revised by replacing the reference to RG 1.163 (Reference 1) with reference to NEI 94-01, Revision 3-A (Reference 2). 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 (3) other outages before the next Type A test, if the interval for the Type A test has been extended to 15 years in accordance with the following sections of NEI 94-01, Revision 3-A:

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

In addition to the inspections performed by the IWE/IWL Containment Inspection Program, procedure ER-AA-380, "Primary Containment Leakrate Testing Program," QCTS 500-14, "Unit 1 IPCRT Engineering Pre-Test Procedure," and QCTS 500-04, "Unit 2 IPCRT Engineering Pre-Test Procedure," require that the structural integrity of the exposed accessible interior and exterior surfaces of the drywell and the containment, including the liner plate, be determined by a visual inspection of those surfaces prior to the Type A Containment Leak Rate Test.

This inspection also fulfills the surveillance requirement of TS SR 3.6.1.1.1 and NEI 94-01.

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

QCNPS Types B and C testing program requires testing of electrical penetrations, airlocks, hatches, flanges, and containment isolation valves in accordance with 10 CFR 50, Appendix J, Option B, and RG 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life. The Types B and C testing program provides a means to protect the health and safety of plant personnel and the public by maintaining leakage from these components below appropriate limits. In accordance with the QCNPS TS 5.5.12, the allowable maximum pathway total Types B and C leakage is 0.6 L_a (Note: For QCNPS, 0.6 L_a is defined as 823.79 scfh and L_a is defined as 1372.99 scfh).

As discussed in NUREG-1493 (Reference 6), 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 QCNPS Type B and Type C test results from 2007 through 2015 for Unit 1 and from 2008 through 2016 for Unit 2 has shown an exceptional amount of margin between the actual As-Found (AF) and As-Left (AL) outage summations and the regulatory requirements. A review of these years As-Found/As-Left test values can be summarized as:

- Unit 1 As-Found minimum pathway leak rate shows an average of 33.4% of 0.6 L_a with a high of 66.1% of 0.6 L_a or 0.3966 of L_a.
- Unit 1 As-Left maximum pathway leak rate shows an average of 39.7% of 0.6 L_a with a high of 41.3% of 0.6 L_a or 0.2480 of L_a.
- Unit 2 As-Found minimum pathway leak rate shows an average of 31.2% of 0.6 L_a with a high of 57.1% of 0.6 L_a or 0.3428 of L_a.
- Unit 2 As-Left maximum pathway leak rate shows an average of 43.2% of 0.6 L_a with a high of 51.6% of 0.6 L_a or 0.3098 of L_a.

Tables 3.5-9 and 3.5-10 provide LLRT data trend summaries for QCNPS Unit 1 since 2007 (last ILRT was 2009) and for Unit 2 since 2008 (last ILRT was 2008).

Table 3.5-9
Unit 1 Types B and C LLRT Combined As-Found/As-Left Trend Summary

Refueling Outage & Year	R19 2007	R20 2009[2]	R21 2011	R22 2013	R23 2015
AF Min Path (scfh)	134.057	229.078	184.265	544.56	283.592[3]
Fraction of L_a[1]	0.0976	0.1668	0.1342	0.3966	0.2066
AL Max Path (scfh)	326.378	336.520	340.523	313.45	321.375
Fraction of L_a	0.2377	0.2451	0.2480	0.2283	0.2341
AL Min Path (scfh)	109.190	155.464	145.136	161.06	175.127
Fraction of L_a	0.0795	0.1132	0.1057	0.1173	0.1276

Note 1: 0.6 L_a = 823.79 scfh and L_a = 1372.99 scfh

Note 2: Q1R20 in 2009 was also an ILRT outage

Note 3: MSIV leakage exceeded individual TS limit (LER 2015-003 written)

Table 3.5-10
Unit 2 Type B and C LLRT Combined As-Found/As-Left Trend Summary

Refueling Outage & Year	R19 2008[2]	R20 2010	R21 2012	R22 2014	R23 2016
AF Min Path (scfh)	106.137	189.452	200.995[3]	470.624	315.866[4]
Fraction of L_a[1]	0.0773	0.1380	0.1464	0.3428	0.2301
AL Max Path (scfh)	238.033	361.436	425.348	370.039	384.262
Fraction of L_a	0.1434	0.2633	0.3098	0.2695	0.2799
AL Min Path (scfh)	93.478	131.765	158.523	102.088	125.432
Fraction of L_a	0.0681	0.0960	0.1155	0.0744	0.0914

Note 1: 0.6 L_a = 823.79 scfh and L_a = 1372.99 scfh

Note 2: Q2R19 in 2008 was also an ILRT outage

Note 3: MSIV leakage exceeded individual TS limit (LER 2012-001 written)

Note 4: MSIV leakage exceeded individual TS limit (LER 2016-001 written)

This summary shows that there has been no As-Found failure that resulted in exceeding the TS 5.5.12 limit of 0.6 L_a and demonstrates a history of successful tests. The As-Found minimum pathway summations represent the high quality of maintenance of Type B and Type C tested components while the As-Left maximum pathway summations represent the effective management of the Containment Leakage Rate Testing Program by the program owner.

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

Tables 3.5-11 and 3.5-12 identify Units 1 and 2 components, respectively, which were on Appendix J, Option B performance-based extended test intervals, but have not demonstrated acceptable performance during the previous two outages. The component test intervals for the components shown have been reduced to 30 months.

Table 3.5-11 Unit 1 Type B and C LLRT Program Implementation Review						
2013-Q1R22						
Component	**As-found scfh**	**Admin Limit Alert/Action scfh**	**As-left scfh**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval**
AOV Gate Valve 1-2001-16 DW Equip. Drain System	295.1	5/10	5.34	Combined leakage w/2001-15. Found valve packing leak.	IR 1488331 CO WO 01624041-6 Adjusted packing, flushed valve. SR frequency change 082070. 47.3 scfh combined w/2001-15 valve	30 month
Check Valve 1-2499-22A Containment Air Monitoring	30.69 (12/2012)	5/10	0.037 (12/2012)	Grit in seat	12/10/2012 PM WO 01397148 replaced valve, IR 1450089 AF failed - grit in seat	30 month
2015-Q1R23						
Component	**As-found scfh**	**Admin Limit Alert/Action scfh**	**As-left scfh**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval**
None						

Table 3.5-12 Unit 2 Type B and C LLRT Program Implementation Review						
2014-Q2R22						
Component	**As-found scfh**	**Admin Limit Alert/Action scfh**	**As-left scfh**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval**
Air Operated Globe 2-0220-44, Primary Sample	31.0	5/10	0.091	Possible seat alignment problem.	Rebuilt, replaced valve trim set. WO 1683615, IR 1643838-02	30 month
Air Operated Plug 2-2001-4 DW Floor Drain System	16.4	5/10	0.021	Normal wear, corrected by adjusting seat.	Rebuild Vlv WO 01728919 , MOD EC 342787 - Replacement of Air Operator WO 00575141 IR 01645048-02 SR 085167	30 month

Table 3.5-12 Unit 2 Type B and C LLRT Program Implementation Review						
Air Operated Valve Gate Vlv 2-2001-15 DW Equip. Drain System (Combined with 2-2001-16)	37.360	5/10	37.360	Accepted leakage into total leakage amount and deferred repair to future outage	EC 24384 is for installing new plug valves (future) WO 99232471 SR 084935 set to 30 month	30 month
Air Operated Valve Gate Vlv 2-2001-16 DW Equip. Drain System (Combined with 2-2001-15)	37.360	5/10	37.360	– Accepted leakage into total leakage amount and deferred repair to future outage	AR 01646791 defer repair to future outage. MD WO 99232471 to replace w/plug valve WO 01730123-01 adjusted packing to original set value. SR 084935	30 month
Check Valve Duo Disc 2-3799-31 RBCCW Supply	Undetermined AF min was 1.079	15/30	0.131	Found a bound butterfly duo disc	WO 01693929-Q2R22C, CAT IDs 1444507 Chamfered Disc, ECR 394799, & 379505 IR 01649483-02	30 month
2016-Q2R23						
Component	**As-found scfh**	**Admin Limit Alert/Action scfh**	**As-left scfh**	**Cause of Failure**	**Corrective Action**	**Scheduled Interval**
Air Operated Globe Valve 2-8804 O2 Analyzer	143.7	5/10	0.067	Appears to be debris on seat	WO1911482 Cleaned up seat, repacked valve, IR 02646542	30 month

The percentage of the total number of QCNPS Appendix J Type B tested components that are on 120-month extended performance-based test intervals is approximately 72% for Unit 1 and 71% for Unit 2.

The percentage of the total number of QCNPS Appendix J Type C tested components that are on 60-month extended performance-based test intervals is approximately 64% for Unit 1 and 67% for Unit 2.

3.5.6 On-Line Monitoring of Primary Containment Atmosphere

During power operation, the primary containment atmosphere is inerted with nitrogen to ensure that oxygen concentration is at or below 4%, by volume. TS 3.6.1.4 requires that drywell pressure be maintained at or below 1.5 psig. Because of this operational requirement, primary containment is typically maintained at an average positive pressure of 1.2 to 1.4 psig.

Primary pressure is continuously indicated and periodically monitored from the Main Control Room. Abnormal high or low drywell pressure is annunciated in the Main Control Room.

Primary containment pressure is periodically monitored in accordance with plant surveillance tests per plant PM. Daily surveillance logs for Modes 1, 2 and 3 include drywell pressure as one of the parameters logged once per shift.

If a primary containment leak were identified, then the TS 3.6.1.A.1 action for an inoperable primary containment would be entered.

3.6 Operating Experience

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

For the QCNPS Primary Containment, the following site specific and industry events have been evaluated for impact:

- Information Notice (IN) 1992-20, "Inadequate Local Leak Rate Testing"
- 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"
- Through-wall Torus Shell Crack at James A. Fitzpatrick Nuclear Power Plant
- GL 87-05, Request for Additional Information - Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells

Each of these areas is discussed in detail in Sections 3.6.1 through 3.6.6, respectively.

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

The issue discussed in IN 1992-20, Inadequate Local Leak Rate Testing, was based on events at four different plants: Quad Cities, Dresden Nuclear Station, Perry Nuclear Plant, and the Clinton Station. 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 QCNPS, the two-ply bellows design was not properly subjected to LLRT pressure and the conclusion of the utility was that the two-ply bellows design could not be Type B LLRT tested as configured. In the events at both Dresden and Perry, flanges were not considered a leakage path when the Type C LLRT test was designed. This omission led to a leakage path that was not discovered until the plant performed an ILRT test.

In the event at Clinton Power Station, relief valve discharge lines that were assumed to terminate below the suppression pool minimum drawdown level were discovered to terminate at a level above that datum. These lines needed to be reconfigured and the valves should have been Type C LLRT tested. To correct this problem, Clinton Power Station removed the vacuum breaker connections and the flanges and extended the pipes to ensure that a water seal would be maintained.

<u>QCNPS Discussion</u>

At QCNPS, LLRT testing of the two-ply stainless steel bellows is performed by a proceduralized series of test techniques, which are; (1) air is first used as the test media to determine leak tightness, (2) followed by helium as a test media if leakage exceeds a predetermined test value, (3) then welding in temporary test fixtures and testing as a Type B component to determine leakage, and (4) then finally, by the replacement of a failed bellow. This test technique was reviewed and supported by the NRC with an exemption granted from testing requirements of Appendix J (Reference 42).

3.6.2 IN 2010-12, "Containment Liner Corrosion"

This IN was issued to alert plant operators to three events that occurred where the steel liner of the containment building was corroded and degraded. At Beaver Valley and Brunswick plants, material was found in the concrete, which trapped moisture against the liner plant 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 concrete 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, an insulating material placed between the concrete floor and the steel liner plate at Salem adsorbed moisture and led to corrosion of the liner plate.

Subsequent to IN 2010-12, the NRC issued Technical Letter Report - Revision 1, "Containment Liner Corrosion Operating Experience Summary," (Reference 19), on August 2, 2011, that summarized this topic across the nuclear industry. The technical letter addresses operating plants that have containment buildings constructed with carbon steel liners in contact with concrete. In the United States, there are 55 pressurized water reactors (PWRs) and 11 BWRs with carbon steel liners in contact with concrete. The focus of the Technical Letter was to evaluate steel containment liner corrosion initiated at the liner/concrete interface.

<u>QCNPS Discussion:</u>

QCNPS was designed and constructed with a Mark I containment that is a freestanding steel primary containment that is not in contact with the concrete (either reinforced steel or prestressed/post-tensioned) containment structure. Because the objective of the Technical Letter is focused on corrosion of steel in contact with concrete, plants with freestanding steel primary containments, (specifically QCNPS, Units 1 and 2) are not included in their review.

QCNPS units have implemented periodic examinations during refueling outages on metallic containment structures or liners in accordance with the Section XI, Subsection IWE. The applicable EGC visual examination procedure requires the conditions described in the Information Notice examples to be recorded. Conditions that may affect containment surface integrity are then required to be evaluated by engineering evaluation or repair/replacement prior to startup from refueling outages.

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

The containment basemat metallic shell and liner plate seam welds of PWRs are embedded in 3-to 4-feet thick concrete floor during construction and are typically covered by a leak-chase channel system that incorporates pressurizing test connections. This system allows for pressure testing of the seam welds for leak-tightness during construction and also 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-inch diameter) that penetrates through the back of the channel and is seal-welded to the channel steel. The tube extends up through the concrete floor slab to a small steel access (junction) box embedded in the floor slab. The steel tube, which may be encased in a pipe, projects up through the bottom of the access box with a threaded coupling connection welded to the top of the tube, allowing for pressurization of the leak-chase channel. After the initial tests, steel threaded plugs or caps are installed in the test tap to seal the leak-chase volume. Gasketed cover plates or countersunk plugs are attached to the top of the access box flush with the containment floor. In some cases, the leak-chase channels with plugged test connections may extend vertically along the circumference of the cylindrical containment shell or liner to a certain height above the floor.

<u>QCNPS Discussion:</u>

No similar deficiencies are present at QCNPS, which is a BWR and does not have a leak-chase channel system inside containment. Containment is periodically inspected as part of the Containment Coatings Program. Water accumulation and corrosion degradation would be observed as part of that program. Nothing significant has been noted and minor corrosion has been promptly repaired.

3.6.4 Through-Wall Torus Shell Crack at James A. Fitzpatrick Nuclear Power Plant

A through-wall torus shell crack was discovered at the James A. Fitzpatrick Nuclear Power Plant (JAF) on June 27, 2005, and was reported via licensee event report (LER) 05-003 (ML052510120). The JAF High Pressure Coolant Injection (HPCI) turbine exhaust line that discharged into the suppression pool is open-ended and does not have an end cap or a sparger.

<u>QCNPS Discussion:</u>

The QCNPS system configurations would not introduce the type of event that occurred at JAF. The HPCI system design does employ the use of a sparger on the turbine exhaust line. Visual VT-2 and VT-3 inspections were performed per the IWE Program on the Torus shell next to the HPCI exhaust penetrations and the support legs to the Torus shell with satisfactory results. No further actions were required.

3.6.5 GL 87-05, "Request for Additional Information – Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells"

GL 87-05 described drywell shell degradation, which occurred at Oyster Creek Nuclear Generating Station as a result of water intrusion into the air gap between the outer drywell surface and the surrounding concrete and subsequent wetting of the sand cushion at the bottom of the air gap.

The cause of this degradation was determined to be from water entering the drywell air gap region, and becoming trapped in the sand cushion region at the base of the air gap. The air gap region surrounds the outside surface of the drywell and extends from the sand cushion region at the bottom, to just below the drywell bellows region at the top. During refueling activities, a potential leakage path could exist through the drywell bellows region, as experienced on the reported Mark I containment. The drywell bellows provides a flexible seal between the drywell and the reactor cavity. The drywell to concrete seal drains are also located in this bellows area. Leakage of these components could allow water to enter the air gap region.

<u>QCNPS Discussion:</u>

In response to NRC Inspection and Enforcement Notice (IEN) 86-99 and GL 87-05, an extensive review was conducted by QCNPS for the potential for drywell steel corrosion in the area of the sand pocket. The response to this GL is contained in UFSAR Section 6.2.1.2.1.2, "Drywell Corrosion Potential," and UFSAR Appendix A, Section A.3.5.2.2, "Degradation Rates of Inaccessible Exterior Drywell Plate Surfaces."

The UFSAR Section 6.2.1.2.1.2 states, in part:

> The QCNPS review included an inspection of the drain lines, initiation of a surveillance program to detect leakage into the annulus, and an evaluation of the actual corrosion rates.
>
> The review concluded that although the potential for degradation of the containment could be postulated to exist, in fact, no corrosion problems were determined to exist. The results of the review determined that: the water present in the sand pocket or inside the drywell was noncorrosive (based on testing), and based on ultrasonic examination, there was no evidence of apparent corrosion.
>
> Also, to ensure active assessment of any future potential problems, surveillance procedures were initiated.

The UFSAR Appendix A, Section A.3.5.2.2 states, in part:

> Commonwealth Edison evaluated the potential effects of corrosion on exterior drywell steel surfaces in the "sand pockets" of Dresden Unit 3 drywell and found that 27 years of service remained before corrosion at the assumed rate would have a significant adverse effect on design basis stresses. The evaluation concluded that the findings were applicable to Dresden Unit 2 and Quad Cities Units 1 and 2 as well.
>
> A program was instituted for the Dresden Unit 3 inaccessible annulus areas to monitor potential corrosion. Dresden Unit 3 is considered the limiting case for potential drywell corrosion among the four Dresden and Quad Cities units.
>
> This inspection at QCNPS is a part of the ASME Section XI, Subsection IWE Program, commitment B.1.26.

A surveillance procedure has been developed and PMIDs generated at QCNPS for monitoring leakage from the Dryer Separator Pit, the Spent Fuel Pool and the Drywell Liner Area Drains (drywell liner). In addition, a separate procedure has been written and PMIDs generated for testing the drain lines for water and to ensure clear, unplugged lines from the sand pocket. The inspection results of the sand pocket and of the drywell liner area for the past 2 refueling outages for each unit is shown in Table 3.6.5-1 below.

Table 3.6.5-1 Unit 1 and Unit 2 Leakage Detection (GL 87-05): (1) Sand Pocket - Procedure QCMPM 1600-03 and (2) Drywell Liner (Dryer Separator Pit, Spent Fuel Pool, Drywell Liner Area Drains) - Procedure QCTS 0820-11				
Refuel	**Date**	**PMID**	**WO**	**Results**
Q1R22 (1) Sand Pocket	January 2013	0189679-02	01556416-01	3 drain lines tested satisfactorily, 1 line (Bay 16) failed (plugged) , addressed by AR01458852 (See Note 1)
Q1R22 (2) Drywell Liner	March 2013	0189679-01	01556404-01	Met Acceptance Criteria – No leakage
Q1R23 (1) Sand Pocket	January 2015	0189679-02	01605627-01	3 drain lines tested satisfactorily, 1 line (Bay 16) failed (plugged), addressed by AR01458852 (See Note 1)
Q1R23 (2) Drywell Liner	March 2015	0189679-01	01623623-01	Met Acceptance Criteria – IR 2463423 written for trending (See Note 2)
Q2R22 (1) Sand Pocket	April 2014	0189680-02	01564143-01	All four drain lines met acceptance criteria - tested satisfactorily.
Q2 R22 (2) Drywell Liner	April 2014	0189680-01	01556403-01	Met acceptance criteria, some hanging drips, addressed by IR 1643923 (See Note 3)
Q2R23 (1) Sand Pocket	March 2016	0189680-02	01758567-01	All four drain lines met acceptance criteria - tested satisfactorily.
Q2 R23 (2) Drywell Liner	March 2016	0189680-01	01728707-01	Did not meet acceptance criteria, addressed by IRs 02644135, 02644136 and 02644137. (See Note 4)

Note 1: AR01458852 - A 2013 surveillance result noted 1 of 4 drain lines was plugged. This is consistent with results reported to NRC in a follow-up letter to NRC related to GL87-05 (dated November 13, 1987). Engineering supports a positon that 1 of 4 plugged drain lines is not significant because if moisture were in the sand pocket it would drain from the other three open drain lines. No additional actions are necessary.

Note 2: IR02463423 - There was no evidence of leakage from the drywell liner area drains, or the sand pocket drains. Leakage is believed to be ground water leakage. The structural integrity of the drywell pedestal is not affected by this issue

Note 3: IR 01643923 - Hanging drips were observed and not seen to fall. Close to trend.

Note 4: IR 02644135 - This leak is under monitoring to test the effectiveness of EC404162. At this time this leakage is expected. No actions are needed in Q2R23. IR02644236 and IR02644137: Water appears to be from ground water leak, is not affecting any equipment in the area and is being controlled by nearby floor drains. There appears to be no concerns with primary containment or the fuel pool from this issue at this time. Closed to information provided and AT02644136-02.

3.7 License Renewal Aging Management

UFSAR Appendix A, "Updated Final Safety Analysis Report (UFSAR) Supplement," contains the UFSAR Supplement as required by 10 CFR 54.21(d) for the QCNPS License Renewal Application (LRA). The NRC issued NUREG-1796, "Safety Evaluation Report Related to the License Renewal of Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2" (Reference 18) that provided their SER of the QCNPS LRA.

The aging management activity descriptions presented in the UFSAR, Appendix A represent commitments for managing aging of the in-scope systems, structures and components during the period of extended operation.

As part of the license renewal effort, it had to be demonstrated that the aging effects applicable for the components and structures within the scope of license renewal would be adequately managed during the period of extended operation.

In many cases, existing activities were found adequate for managing aging effects during the period of extended operation. In some cases, aging management reviews revealed that existing activities required enhancement to adequately manage applicable aging effects. In a few cases, new activities were developed to provide added assurance that aging effects are adequately managed.

The following programs/activities are credited with the aging management of the Primary Containment (Drywell and Torus).

- <u>10 CFR 50, Appendix J (Supplement Appendix A.1.28)</u>

  The 10 CFR 50, Appendix J aging management program monitors leakage rates through the containment pressure boundary, including the drywell and torus, penetrations, fittings, and other access openings; in order to detect degradation of containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. The Appendix J program also manages changes in material properties of gaskets, O-rings, and packing materials for the containment pressure boundary access points. The containment leak rate tests are performed in accordance with the regulations and guidance provided in 10 CFR 50, Appendix J, Option B, RG 1.163, "Performance-Based Containment Leak-Testing Program," NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J," and ANSI/ANS 56.8, "Containment System Leakage Testing Requirements."

- <u>ASME Section XI, Subsection IWE (Supplement Appendix A.1.26)</u>

  The ASME Section XI, Subsection IWE aging management program consists of periodic visual examination for signs of degradation, and limited surface or volumetric examination when augmented examination is required. The program covers steel containment shells and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The program includes assessment of damage and corrective actions. The program complies with

ASME Section XI, Subsection IWE for steel containments (Class MC), 1992 Edition including 1992 Addenda.

NOTE: The Second CISI Interval was updated for QCNPS, Units 1 and 2, with effective dates of September 9, 2008, through September 8, 2018. The ASME Section XI Code of Record for the Second CISI Interval is the 2001 Edition through the 2003 Addenda.

- Protective Coating Monitoring and Maintenance Program (Supplement Appendix A.1.32)

  The protective coating monitoring and maintenance aging management program consists of guidance for selection, application, inspection, and maintenance of Service Level I protective coatings. This program is implemented in accordance with RG 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants," Revision 0; ANSI N101 4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities;" and, the guidance of EPRI TR-109937, "Guidelines on Nuclear Safety-Related Coating." Prior to the period of extended operation, the program will be revised to include thorough visual inspection of Service Level I coatings near sumps or screens for the ECCS, preinspection review of previous reports so that trends can be identified, and analysis of suspected causes of any coating failures.

  NOTE: The program to maintain containment coatings was developed to meet the requirements of RG 1.54, Revision 0. This program is implemented at QCNPS with procedures ER-AA-330-008, "Exelon Safety-Related (Service Level I) Protective Coatings," and ER-QC-330-1000, "Primary Containment and Coating Inspections."

3.8 NRC SE Limitations and Conditions

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

The NRC 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 were satisfied:

Table 3.9-1
NEI 94-01 Revision 2-A Limitations and Conditions

Limitation/Condition (From Section 4.0 of SE)	QCNPS 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.)	QCNPS will utilize the definition in NEI 94-01 Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.
The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3.)	Reference Section 3.5.2 (Tables 3.5-4 and 3.5-5) of this LAR submittal.
The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3.)	Reference Section 3.5.2 (Tables 3.5-4 and 3.5-5) of this LAR 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 to the containment structure. Modification is underway to comply with NRC Order EA-13-109, to install a hardened containment vent system (does not directly modify containment). This NRC Order is the result of the Fukushima Dai-ichi event. See Section 3.1.7 of this LAR submittal for additional details. (Note: Work on the hardened containment vent modification is currently on hold due to other licensing actions. Upon installation, this modification will be tested and maintained in accordance with the Appendix J and CISI Programs as applicable.
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.)	QCNPS will follow the requirements of NEI 94-01 Revision 3-A, Section 9.1. This requirement has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01. In accordance with the requirements of 94-01 Revision 2-A, SE Section 3.1.1.2, QCNPS will also demonstrate to the NRC that an unforeseen emergent condition exists in the event an extension beyond the 15-year interval is required.
For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past containment ILRT data.	Not applicable. QCNPS was not licensed under 10 CFR Part 52.

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

The NRC found that the guidance in NEI TR 94-01, Revision 3, was acceptable for referencing by licensees in the implementation of the optional performance-based requirements of Option B to 10 CFR 50, Appendix J. However, the NRC identified two conditions on the use of NEI TR 94-01, Revision 3 (Reference NEI 94-01 Revision 3-A, NRC SE 4.0, Limitations and Conditions):

Topical Report Condition 1

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

Response to Condition 1

Condition 1 presents the following three (3) separate issues that are required to be addressed:

- ISSUE 1 – The allowance of an extended interval for Type C LLRTs of 75 months carries the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit.
- ISSUE 2 – In addition, a corrective action plan shall be developed to restore the margin to an acceptable level.
- ISSUE 3 – Use of the allowed 9-month extension for eligible Type C valves is only authorized for non-routine emergent conditions with exceptions as detailed in NEI 94-01, Revision 3-A, Section 10.1.

Response to Condition 1, ISSUE 1

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

Response to Condition 1, ISSUE 2

When the potential leakage understatement adjusted Types B and C MNPLR total is greater than the QCNPS, Units 1 and 2, leakage summation limit of 0.5 L_a, but less than the regulatory limit of 0.6 L_a, then an analysis and determination of a corrective action plan shall be prepared to restore the leakage summation margin to less than the QCNPS leakage limit. The corrective action plan shall focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues so as to maintain an acceptable level of margin.

Response to Condition 1, ISSUE 3

QCNPS, Units 1 and 2 will apply the 9-month allowable interval extension period only to eligible Type C components and only for non-routine emergent conditions. Such occurrences will be documented in the record of tests.

<u>*Topical Report Condition 2*</u>

The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time. The containment leakage condition monitoring regime involves a portion of the penetrations being tested each refueling outage, nearly all LLRTs being performed during plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leakage rates for the just tested penetrations are summed with the as-left minimum pathway leakage rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves, which in the aggregate, will show increasing leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate maintenance may extend this increasing leakage potential. Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total is used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI TR 94-01, Revision 3, Section 12.1.

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

Response to Condition 2

Condition 2 presents the following two (2) separate issues that are required to be addressed:

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

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

Response to Condition 2, ISSUE 1

The change in going from a 60-month extended test interval for Type C tested components to a 75-month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25% in the LLRT periodicity. As such, QCNPS, Units 1 and 2 will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the actual As-Left leak rate, which will increase the As-Left leakage total for each Type C component currently on greater than a 60-month test interval up to the 75-month extended test interval. This will result in a combined conservative Type C total for all 75-month LLRTs being "carried forward" and will be included whenever the total leakage summation is required to be updated (either while on-line or following an outage).

When the potential leakage understatement adjusted leak rate total for those Type C components being tested on greater than a 60-month test interval up to the 75-month extended test interval is summed with the non-adjusted total of those Type C components being tested at less than or equal to a 60-month test interval, and the total of the Type B tested components, results in the MNPLR being greater than the QCNPS leakage summation limit of 0.50 L_a, but less than the regulatory limit of 0.6 L_a, then an analysis and corrective action plan shall be prepared to restore the leakage summation value to less than the QCNPS leakage limit. The corrective action plan should focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues (Reference 44).

Response to Condition 2, ISSUE 2

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

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

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

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

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

3.9 Conclusions

NEI 94-01, Revision 3-A, dated July 2012, and the limitations and conditions specified in NEI 94-01, Revision 2-A, dated October 2008, describe an NRC-accepted approach for implementing the performance-based requirements of 10 CFR 50, Appendix J, Option B. It incorporated the regulatory positions stated in RG 1.163 and includes provisions for extending Type A intervals to 15 years and Type C test intervals to 75 months. NEI 94-01, Revision 3-A delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance test frequencies. QCNPS is adopting the guidance of NEI 94-01, Revision 3-A, and the limitations and conditions specified in NEI 94-01, Revision 2-A, for the QCNPS, Units 1 and 2, 10 CFR 50, Appendix J testing program plan.

Based on the previous ILRTs conducted at QCNPS, Units 1 and 2, 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, drywell Inspections and the overlapping inspection activities performed as part of the following QCNPS inspection programs:

- Containment Inservice Inspection Program (IWE)
- Containment Inspections per TS SR 3.6.1.1.1
- Containment Coatings Inspection and Assessment Program

This experience is supplemented by risk analysis studies, including the QCNPS risk analysis provided in Attachment 3. 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 QCNPS 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 50.54(o) requires primary reactor containments for water-cooled power reactors to be subject to the requirements of Appendix J to 10 CFR 50, "Leakage Rate Testing of Containment of Water Cooled Nuclear Power Plants." 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, Type B, and Type C 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. Similarly, the proposed change to the Type C test frequencies will not directly result in an increase in containment leakage.

EPRI TR-1009325, Revision 2-A (Reference 20), 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 3-A, Section 9.2.3.1 (Reference 2), 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 2-A), 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 reviewed NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2. For NEI TR 94-01, Revision 2, the NRC determined that it described an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J. This guidance includes provisions for extending Type A ILRT intervals up to 15 years and incorporates the regulatory positions stated in RG 1.163. The NRC finds that the Type A testing methodology, as described in ANSI/ANS-56.8-2002, and the modified testing frequencies recommended by NEI TR 94-01, Revision 2, 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 2, a risk-informed methodology using plant-specific risk insights and industry ILRT performance data to revise ILRT surveillance frequencies, the NRC 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 and RG 1.174. The NRC, 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 SE.

The NRC reviewed NEI TR 94-01, Revision 3, and determined that it described an acceptable approach for implementing the optional performance-based requirements of Option B to 10 CFR 50, Appendix J, as modified by the limitations and conditions summarized in Section 4.0 of the associated SE. This guidance included provisions for extending Type C LLRT intervals up to 75 months. Type C testing ensures that individual CIVs are essentially leak tight. In addition, aggregate Type C leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths. The NRC, therefore, found that this guidance, as modified to include two limitations and conditions, was acceptable for referencing by licensees proposing to amend their TS in regards to containment leakage rate testing. Any applicant may reference NEI TR 94-01, Revision 3, as modified by the associated SE and approved by the NRC, and the limitations and conditions specified in NEI 94-01, Revision 2-A, dated October 2008, 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 to extend the Type A Test Frequency to 15 years and the Type C test frequency to 75 months as previously authorized by the NRC:

- Surry Power Station, Units 1 and 2 (Reference 24)
- Donald C. Cook Nuclear Plant, Units 1 and 2 (Reference 25)
- Beaver Valley Power Station, Unit Nos. 1 and 2 (Reference 26)
- Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (Reference 27)
- Peach Bottom Atomic Power Station, Units 2 and 3 (Reference 28)
- Comanche Peak Nuclear Power Plant, Units 1 and 2 (Reference 39)

4.3 No Significant Hazards Consideration

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 Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively. The proposed change revises Technical Specifications (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for the permanent extension of the Type A Integrated Leak Rate Testing (ILRT) and Type C Leak Rate Testing frequencies.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

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

   Response: No

   The proposed activity involves revision of the Quad Cities Nuclear Power Station (QCNPS) Technical Specification (TS) 5.5.12, Primary Containment Leakage Rate Testing Program, to allow the extension of the QCNPS, Units 1 and 2, Type A containment integrated leakage rate test interval to 15 years, and the extension of the Type C local leakage rate test interval to 75 months. The current Type A test interval of 120 months (10 years) would be extended on a permanent basis to no longer than 15 years from the last Type A test. The existing Type C test interval of 60 months for selected components would be extended on a performance basis to no longer than 75 months. Extensions of up to nine months (total maximum interval of 84 months for Type C tests) are permissible only for non-routine emergent conditions.

   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. 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 dose risk for changing the Type A Integrated Leak Rate Test (ILRT) interval from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences for QCNPS, is 1.0E-02 person-rem/yr (0.31%) using the Electric Power Research Institute (EPRI) guidance with the base case corrosion included. The change in dose risk drops to 2.7E-03 person-rem/yr (0.08%) when using the EPRI Expert Elicitation methodology. The values calculated per the EPRI guidance are all lower than the acceptance criteria of less than or equal to 1.0 person-rem/yr or less than 1.0% person-rem/yr defined in Section 1.3 of Attachment 3 to this LAR). Therefore, this proposed extension does not involve a significant increase in the probability of an accident previously evaluated.

As documented in NUREG-1493, "Performance-Based Containment Leak-Test Program," dated January 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 QCNPS, Units 1 and 2 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) Section XI, 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. Based on the above, the proposed test interval extensions do not significantly increase the consequences of an accident previously evaluated.

The proposed amendment also deletes an exception previously granted in amendments 220 and 214 to allow one-time extensions of the ILRT test frequency for QCNPS, Units 1 and 2, respectively. This exception was for an activity that has already taken place; therefore, this deletion is solely an administrative action that does not result in any change in how QCNPS, Units 1 and 2 are operated.

Therefore, the proposed change does not involve 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 TS 5.5.12, "Primary Containment Leakage Rate Testing Program," involves the extension of the QCNPS, Units 1 and 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months. 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.

   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.

   The proposed amendment also deletes an exception previously granted under TS Amendments 220 and 214 to allow the one-time extension of the ILRT test frequency for

QCNPS, Units 1 and 2, respectively. This exception was for an activity that has already taken place; therefore, this deletion is solely an administrative action that does not result in any change in how the QCNPS units are operated.

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

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

Response: No

The proposed amendment to TS 5.5.12 involves the extension of the QCNPS, Units 1 and 2 Type A containment test interval to 15 years and the extension of the Type C test interval to 75 months for selected components. 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 and Type C tests for QCNPS, Units 1 and 2. The proposed surveillance interval extension is bounded by the 15-year ILRT interval and the 75-month Type C test interval currently authorized within NEI 94-01, Revision 3-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 ASME Section XI and 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 and Type C test intervals.

The proposed amendment also deletes exceptions previously granted to allow one-time extensions of the ILRT test frequency for QCNPS, Units 1 and 2. This exception was for an activity that has taken place; therefore, the deletion is solely an administrative action and does not change how QCNPS is operated and maintained. Thus, there is no reduction in any margin of safety.

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

Based on the above evaluation, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (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 the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

EGC 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, "Standards for Protection Against Radiation." 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, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

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

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

3. Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated May 2011

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

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

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

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

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

9. Letter from M. J. Maxin (NRC) to J. C. Butler (NEI), "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 (ML081140105)

10. 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 Part 50, Appendix J,' (TAC No. ME2164)," dated June 8, 2012 (ML121030286)

11. EPRI TR-1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325," dated October 2008

12. Quad Cities Calculation QDC-1600-M-1617, "Determination of the Area of Primary Containment After Construction," dated January 24, 2008 [Calculation that determined 80% of containment pressure boundary surface area is accessible]

13. Letter from J. F. Stang (NRC) to D. L. Farrer (Commonwealth Edison Company), "Issuance of Amendments [No. 169 and 165] Related to 10 CFR Part 50, Appendix J, Option B (TAC Nos. M94061, M94062, M94065 and M94066)," dated January 11, 1996 (ML021160123)

14. Letter from S. N. Bailey (NRC) to O. D. Kingsley (Commonwealth Edison Company), "Issuance of Amendments (TAC Nos. MA6082 and MA6083)," Enclosure 3 - Safety Evaluation Related to Amendments No 192 and No 188, dated December 21, 1999 (ML993630246 and ML993630259)

15. Letter from S. N. Bailey (NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0842 and MB0843)," Enclosure 3 - Safety Evaluation Related to Amendments No 202 and No 198, dated December 21, 2001 (ML013530380 and ML013540222)

16. Letter from L. W. Rossbach (NRC) to J. L Skolds (Exelon Generation Company, LLC), "Issuance of Amendment - Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2, Excess Flow Check Valves (TAC Nos. MB7732, MB7733, MB7734, AND MB7735)," dated October 10, 2003 (ML032740364)

17. Letter from L. W. Rossbach (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments [Nos. 220 and 214] Regarding Containment Leakage Rate Testing (TAC Nos. MB7861 and MB7862)," dated March 8, 2004 (ML040280368)

18. Safety Evaluation Report, Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2, (NUREG-1796), dated October 28, 2004 (ML043060582)

19. Technical Letter Report, Revision 1, "Containment Liner Corrosion Operating Experience Summary," by D. S. Dunn, A. L. Pulvirenti and M. A. Hiser, issued by the NRC on August 2, 2011 (ML112070867)

20. Electric Power Research Institute, Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325, EPRI TR-1018243, dated October 2008

21. Letter from B. Rybak (Commonwealth Edison Company) to H. R. Denton (NRC), dated June 27, 1983, "Quad Cities Nuclear Power Station Units 1 and 2 Plant Unique Analysis Report," Revision 0, dated May 1983, UFSAR, Section 3.8.2, Reference 1. (Reference described on Page 3.8-28)

22. Letter from J. A. Zwolinski (NRC) to D. L. Farrar (Commonwealth Edison Company), February 15, 1986, "Mark I Containment Long Term Program," UFSAR Section 3.8.2, Reference 2 (Reference described on Page 3.8-28)

23. NEI 00-02, "Probabilistic Risk Assessment Peer Review Process Guidance," Rev. A3, dated March 2000

24. Letter to D. Heacock from S. Williams (NRC), Surry Power Station, "Units 1 and 2 - Issuance of Amendment Regarding the Containment Type A and Type C Leak Rate Tests," dated July 3, 2014 (ML14148A235)

25. Letter to L. Weber from A. Dietrich (NRC), "Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments RE: Containment Leakage Rate Testing Program," dated March 30, 2015 (ML15072A264)

26. Letter to E. Larson from T. Lamb (NRC), "Beaver Valley Power Station, Unit Nos. 1 and 2 - Issuance of Amendment Re: License Amendment Request to Extend Containment Leakage Rate Test Frequency," dated April 8, 2015 (ML15078A058)

27. Letter to G. Gellrich from A. Chereskin (NRC), "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendments Re: Extension of Containment Leakage Rate Testing Frequency," dated July 16, 2015 (ML15154A661)

28. Letter to B. Hanson from R. Ennis (NRC), "Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Extension of Type A and Type C Leak Rate Test Frequencies (TAC Nos. MF5172 and MF5173)," dated September 8, 2015 (ML15196A559)

29. American Society of Mechanical Engineers, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME RA-S-2002, New York, New York, April 2002

30. ASME/American Nuclear Society, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa-2009, dated March 2009. Addendum A to RA-S-2008

31. Quad Cities Nuclear Power Station PRA Peer Review Report, BWROG Final Report, dated February 2000

32. Letter from Mr. C. H. Cruse (Constellation Nuclear, Calvert Cliffs Nuclear Power Plant) to NRC, "Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension," dated March 27, 2002 (ML020920100)

33. Boiling Water Reactors Owners' Group, BWROG PSA Peer Review Certification Implementation Guidelines, Revision 3, dated January 1997

34. Reactor Oversight Program MSPI Bases Document, Quad Cities Generating Station, Revision 5b, dated December 15, 2011

35. Quad Cities Nuclear Power Station PRA Peer Review Report (Internal Flooding) Using ASME PRA Standards, dated November 2010

36. QC-PSA-13 Self-Assessment of the Quad Cities PRA Against the Combined ASME/ANS PRA Standard Requirements, Revision 0

37. QCNPS UFSAR Section 3.8, Figure 3.8-39, "Typical Electrical Penetration Assembly Canister"

38. QC-PSA-16, Self-Assessment of the Quad Cities PRA Against the Combined ASME/ANS PRA Standard Requirements Revision 5

39. Letter from B. Singal (NRC) to R. Flores (Luminant Generation Co.), "Comanche Peak Nuclear Power Plant, Units 1 and 2 – Issuance of Amendments Re: Technical Specification Change for Extension of the Integrated Leak Rate Test Frequency from 10 to 15 Years (CAC Nos. MF5621 and MF5622)," dated December 30, 2015 (ML15309A073)

40. Letter from Maitri Banerjee (NRC) to C.M. Crane (Exelon Generation Company, LLC) pertaining to Issuance of Amendments RE: Adoption of Alternative Source Term Methodology. Letter contained Safety Evaluation Report for Amendments No. 233 (Unit 1) and No. 229 (Unit 2), dated September 11, 2006 (ML062070290). Subsequent letter from Daniel S. Collins (NRC) to C. M. Crane (Exelon Nuclear, EGC) revising values on pages 9, 11 and 12 of SE. Changes have no impact on Appendix J and this LAR application (ML062680404)

41. Letter from D. B. Vassallo (NRC) to D. L. Farrar (Commonwealth Edison Company), "Re: Quad Cities Nuclear Power Station, Units 1 and 2," [Exemption from 10 CFR 50.54(o) and Appendix J pertaining to test sequence for Type A and C tests, the exclusion of instrument line and MSIVs for the Type C test requirement and extends the interval between Type B tests for the containment airlock], dated June 12, 1984 (ML020930631)

42. Letter from B. A. Boger (NRC) to T. J. Kovach (Commonwealth Edison Company), "Exemption from the Testing Requirements of Appendix J to 10 CFR Part 50 for Dresden and Quad Cities Nuclear Power Stations (TAC Nos. M81299, M81300, M81301, and M81302)," dated February 6, 1992. (ML021150419) [Subsequent letter dated February 9, 1995, from the NRC further revised imposed test measures (TAC Nos. M90628, M90629, M90630, and M90631)]

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

44. Letter from K. Mulligan (Entergy Operations, Inc.) to NRC, "Grand Gulf Nuclear Station Response to Request for Additional Information Regarding License Amendment Request to Revise Technical Specification for Containment Leak Rate Testing, Grand Gulf Nuclear Station, Unit 1, Docket No. 50-416, License No. NPF-29," Entergy document GNRO-2015/00063 (ML15302A042)

45. Calculation: "Torus Pitting Corrosion Acceptance Criteria for Quad Cities Nuclear Power Station Units 1 and 2," Calculation File No: 64.305.2029, Project No: 1598.0015, Revision 1, dated May 6, 1994

46. NEI 05-04, Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard, Revision 2, dated November 2008

47. Quad Cities Nuclear Generation Station PRA Peer Review Report (All applicable SRs except Internal Flooding), April 2017

ATTACHMENT 2
Markup of Proposed Technical Specifications Pages

Quad Cities Nuclear Power Station, Units 1 and 2

Renewed Facility Operating License Nos. DPR-29 and DPR-30

REVISED TECHNICAL SPECIFICATIONS PAGES

5.5-11
5.5-12

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

b. A loss of safety function exists when, assuming no concurrent single failure, and assuming no concurrent loss of offsite power or loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

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

3. A required system redundant to support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. 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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008.

5.5.12 Primary Containment Leakage Rate Testing Program

a. This program shall establish the leakage 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 ~~exemption~~ exemptions. This program shall be in accordance with the guidelines contained in ~~Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995, as modified by the following exceptions:~~

~~1. NEI 94-01 - 1995, Section 9.2.3: The first Unit 1 Type A test performed after the July 23, 1994, Type A test shall be performed no later than July 22, 2009.~~

(continued)

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

~~2. NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after the May 17, 1993, Type A test shall be performed no later than May 16, 2008.~~

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 43.9 psig.

c. The maximum allowable primary containment leakage rate, L_a, at P_a, is 3% of primary containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Primary containment overall leakage rate acceptance criterion is ≤ 1.0 La. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for the combined Type B and Type C tests, and ≤ 0.75 La for Type A tests.

2. Air lock testing acceptance criteria is the overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa.

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

5.5.13 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation (CREV) System, CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposure in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive

(continued)

ATTACHMENT 3

QC-LAR-03, "Risk Assessment for QCNPS Regarding the ILRT (Type A) Permanent Extension Request"

Exelon Generation.

RISK MANAGEMENT TEAM

RM DOCUMENTATION NO. QC-LAR-03	**REV: 2**	**PAGE NO. 1**

STATION: Quad Cities Nuclear Power Station (QCNPS)
UNIT(s) AFFECTED: 1 & 2

TITLE: Risk Assessment for QCNPS Regarding the ILRT (Type A) Permanent Extension Request

SUMMARY:
QCNPS is pursuing a License Amendment Request (LAR) to permanently extend the 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 QCNPS Unit 1 and Unit 2 containment ILRT interval to 15 years. Rev. 1 changed the external events multiplier (section 5.7), leakage rate multiplier (section 4.2) and editorial changes. Rev. 2 revises the PRA Quality Section to incorporate the 2017 QC FPIE peer review.

This is a Category I Risk Management Document in accordance with ER-AA-600-1012 Risk Management Documentation [24], which requires independent review and approval, ER-AA-600-1046 Risk Metrics – NOED and LAR [25] and ER-AA-600-1051, "Risk Assessment of Surveillance Test Frequency Changes" [26].

[ ] Review required after periodic update

[X] Internal RM Documentation **[ ] External RM Documentation**

Electronic Calculation Data Files:
Microsoft Excel QC_ILRT-Final Rev 1.xlsx, 07/06/2016, 1:06 AM, 266 KB

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

This RM documentation supersedes: QC-LAR-03 Rev. 1

	Print	Sign	Date
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Reviewed by:	Don Vanover	Donald E. Vanover	4/5/17
Approved by:	Eugene M. Kelly	Eugene M. Kelly	4/6/17

TABLE OF CONTENTS

APPENDICIES

1.0 OVERVIEW

The risk assessment associated with implementing a permanent extension of the QCNPS Unit 1 and Unit 2 Integrated Leak Rate Test (ILRT) interval to 15 years is described in this document.

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk associated with implementing a permanent extension of the QCNPS Units 1 and 2 containment Type A ILRT interval from ten years to fifteen years. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology outlined in Electric Power Research Institute (EPRI) TR-104285 [2], as updated by the EPRI Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals (EPRI TR-1018243) [3], the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a plant's licensing basis as outlined in Regulatory Guide (RG) 1.174 [4], and the methodology used for Calvert Cliffs to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval [5]. The format of this document is consistent with the intent of the Risk Impact Assessment Template for evaluating extended integrated leak rate testing intervals provided in the EPRI TR-1018243 [3].

1.2 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than the normal containment leakage of 1.0 La (allowable leakage).

The basis for a 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [6], "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 assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI TR-104285 [2].

The NRC report on performance-based leak testing, NUREG-1493 [6], analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined for a comparable BWR plant, that increasing the containment leak rate from the nominal 0.5 percent per day to 5 percent per day leads to a barely perceptible increase in total population exposure, and increasing the leak rate to 50 percent per day increases the total population exposure by less than 1 percent. Because ILRTs represent substantial resource expenditures, it is desirable to show that extending the ILRT interval will not lead to a substantial increase in risk from containment isolation failures to support a reduction in the test frequency for QCNPS. The current analysis is being performed to confirm these conclusions based on QCNPS specific PRA models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 [2] methodology to perform the risk assessment. In October 2008, EPRI 1018243 [3] was issued to develop a generic methodology for the risk impact assessment for ILRT interval extensions to 15 years using current performance data and risk informed guidance, primarily NRC Regulatory Guide 1.174 [4]. This more recent EPRI document considers the change in population dose, large early release frequency (LERF), and containment conditional failure probability (CCFP), whereas EPRI TR-104285 considered only the change in risk based on the change in population dose. This ILRT interval extension risk assessment for QCNPS Unit 1 and QCNPS Unit 2 employs the

EPRI 1018243 methodology, with the affected System, Structure, or Component (SSC) being the primary containment boundary.

1.3 ACCEPTANCE CRITERIA

The acceptance guidelines in RG 1.174 [4] 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 1.0E-06 per reactor year and increases in large early release frequency (LERF) less than 1.0E-07 per reactor year. Note that a separate discussion in Section 5.8 of this risk assessment confirms that the CDF is negligibly impacted by the proposed ILRT interval change for QCNPS. Therefore, since the Type A test has only a minimal impact on CDF for QCNPS, the relevant criterion is the change in LERF. RG 1.174 also defines “small” changes in LERF as below 1.0E-06 per reactor year, provided that the total LERF from all contributors (including external events) can be reasonably shown to be less than 1.0E-05 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) is also calculated to help ensure that the defense-in-depth philosophy is maintained.

With regard to population dose, examinations of NUREG-1493 and Safety Evaluation Reports (SERs) for one-time interval extension (summarized in Appendix G of [3]) indicate a range of incremental increases in population dose(1) that have been accepted by the NRC. The range of incremental population dose increases is from ≤0.01 to 0.2 person-rem/yr and 0.002 to 0.46% of the total accident dose. The total doses for the spectrum of all accidents (Figure 7-2 of NUREG-1493) result in health effects that are at least two orders of magnitude less than the NRC Safety Goal Risk. Given these perspectives, the NRC SER on this issue [7] defines a “small” increase in population

(1) The one-time extensions assumed a large leak (EPRI class 3b) magnitude of 35 La, whereas this analysis uses 100 La.

dose as an increase of ≤ 1.0 person-rem per year, or ≤1 % of the total population dose (when compared against the baseline interval of 3 tests per 10 years), whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This definition has been adopted for the QCNPS analysis.

The acceptance criteria are summarized below.

1. The estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years must be demonstrated to be "small." (Note that Regulatory Guide 1.174 defines "very small" changes in risk as increases in CDF less than 1.0E-06 per reactor year and increases in LERF less than 1.0E-07 per reactor year. Since the type A ILRT test does not have a significant impact on CDF for QCNPS, the relevant risk metric is the change in LERF. Regulatory Guide 1.174 also defines "small" risk increase as a change in LERF of less than 1.0E-06 reactor year.) Therefore, a small change in risk for this application is defined as a LERF increase of less than 1.0E-06.
2. Per the NRC SE, a small increase in population dose is also 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.
3. In addition, the NRC SE notes that a small increase in Conditional Containment Failure Probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests (typically about 1% or less, with the largest increase being 1.2%). This would require that the increase in CCFP be less than or equal to 1.5 percentage points.

2.0 METHODOLOGY

A simplified bounding analysis approach consistent with the EPRI methodology [3] is used for evaluating the change in risk associated with increasing the test interval to fifteen years. The analysis uses results from a Level 2 analysis of core damage scenarios from the current QCNPS PRA models of record [16, 17] and the subsequent containment responses to establish the various fission product release categories including the release size.

The six general steps of this assessment are as follows:

1. Quantify the baseline risk in terms of the frequency of events (per reactor year) for each of the eight containment release scenario types identified in the EPRI report [3].
2. Develop plant-specific population dose rates (person-rem per reactor year) for each of the eight containment release scenario types from plant specific consequence analyses.
3. Evaluate the risk impact (i.e., the change in containment release scenario type frequency and population dose) of extending the ILRT interval to fifteen years.
4. Determine the change in risk in terms of Large Early Release Frequency (LERF) in accordance with RG 1.174 and compare this change with the acceptance guidelines of RG 1.174 [4].
5. Determine the impact on the Conditional Containment Failure Probability (CCFP)
6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis and to variations in the fractional contributions of large isolation failures (due to liner breach) to LERF.

Furthermore,

- Consistent with the previous industry containment leak risk assessments, the QCNPS assessment uses population dose as one of the risk measures. The other risk measures used in the QCNPS assessment are the conditional containment failure probability (CCFP) for defense-in-depth considerations, and change in LERF to demonstrate that the acceptance guidelines from RG 1.174 are met.

- This evaluation for QCNPS uses ground rules and methods to calculate changes in the above risk metrics that are consistent with those outlined in the current EPRI methodology [3].

3.0 GROUND RULES

The following ground rules are used in the analysis:

- The QCNPS Level 1 and Level 2 internal events PRA models provide representative core damage frequency and release category frequency distributions to be utilized in this analysis. The technical adequacy of the PRA models is consistent with the requirements of Regulatory Guide 1.200 as relevant to this ILRT risk assessment. PRA adequacy is discussed in Appendix A.
- It is appropriate to use the QCNPS internal events PRA model as a gauge to effectively describe the risk change attributable to the ILRT extension. It is reasonable to assume that the impact from the ILRT extension (with respect to percent increases in population dose) will not substantially differ if external events were to be included in the calculations; however, external events have been accounted for in the analysis based on the available information for QCNPS.
- Dose results for the containment failures modeled in the PRA can be characterized by information provided in License Renewal Environmental Report for Quad Cities Nuclear Power Station, Units 1 and 2 [8]. The Severe Accident Mitigation Alternatives (SAMA) analysis in the Environmental Report used a population estimated for the year 2032 and is judged reasonable for use in this ILRT evaluation. The current QCNPS power level of 2957 MWth as documented in the Updated Safety Analysis Report (USAR) [27] was the power level used as input to the License Renewal Environmental Report L3 PSA as documented in Section F.3.5 of the Environmental Report. Therefore, no correction for power level is required.
- Accident classes describing radionuclide release end states and their definitions are consistent with the EPRI methodology [3] and are summarized in Section 4.2.
- The representative containment leakage for Class 1 sequences is 1 La. Class 3 accounts for increased leakage due to Type A inspection failures.
- The representative containment leakage for Class 3a is 10 La and for Class 3b sequences is 100 La, based on the recommendations in the latest EPRI report [3] and as recommended in the NRC SE on this topic [7]. 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.
- Based on the EPRI methodology and the NRC SE, the Class 3b sequences are categorized as LERF and the increase in Class 3b sequences is used as a surrogate for the ΔLERF metric.

- 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 on the conclusions from this analysis will result from this separate categorization.
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.
- The use of the estimated 2030 population data is appropriate for this analysis. Precise evaluations of the projected population would not significantly impact the quantitative results, nor would it change the conclusions.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [9].

4.0 INPUTS

This section summarizes the following:

- Section 4.1 - General resources available as input
- Section 4.2 - Plant specific resources required
- Section 4.3 - Impact of extension on detection of component failures that lead to leakage (small and large)
- Section 4.4 - Impact of extension on detection of steel liner corrosion that leads to leakage

4.1 GENERAL RESOURCES AVAILABLE

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

1. NUREG/CR-3539 [10]
2. NUREG/CR-4220 [11]
3. NUREG-1273 [12]
4. NUREG/CR-4330 [13]
5. EPRI TR-105189 [9]
6. NUREG-1493 [6]
7. EPRI TR-104285 [2]
8. Calvert Cliffs liner corrosion analysis [5]
9. EPRI 1018243 [3]
10. NRC Final Safety Evaluation [7]

The 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 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 LLRT test intervals on at-power public risk. The eighth study addresses the impact of age-related degradation of the containment liners on ILRT evaluations. EPRI 1018243 complements the previous EPRI report and provides the results of an expert elicitation process to determine the relationship between pre-existing containment leakage probability and magnitude. Finally, the NRC Safety Evaluation (SE) documents the acceptance by the NRC of the proposed methodology with a few exceptions. These exceptions (associated with the ILRT Type A tests) were addressed in the Revision 2-A of NEI 94-01 (and maintained in Revision 3-A of NEI 94-01) and the final version of the updated EPRI report [3], which was used for this application.

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

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [14] 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 [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. It assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 events in 740 reactor years and conservatively assuming a one-year duration for each event.

NUREG-1273 [12]

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

NUREG/CR-4330 [13]

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

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

EPRI TR-105189 [9]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed 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 result of the study concluded that a small but measurable safety benefit (shutdown CDF reduced by 1.0E-8/yr to 1.0E-7/yr) is realized from extending the test intervals from 3 per 10 years to 1 per 10 years.

NUREG-1493 [6]

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

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending Integrated Leak Rate Test (ILRT) and (Local Leak Rate Test) LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 [15] Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 [6] in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight categories of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
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 failure due to core damage accident phenomena
8. Containment bypass

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

> *"These study results show that 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..."*

Release Category Definitions

The EPRI methodology [2, 3] defines the accident classes that may be used in the ILRT extension evaluation. These containment failure classes, reproduced in Table 4.1-1, are used in this analysis to determine the risk impact of extending the Containment Type A test interval as described in Section 5 of this report.

TABLE 4.1-1

EPRI [2] / NEI CONTAINMENT FAILURE CLASSIFICATIONS

CLASS	DESCRIPTION
1	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined by the maximum allowable leakage rate values L_a, under Appendix J for that plant
2	Containment isolation failures (as reported in the IPEs) include those accidents in which there is a failure to isolate the containment.
3	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal (i.e., provide a leak-tight containment) is not dependent on the sequence in progress.
4	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 3 isolation failures, but is applicable to sequences involving Type B tests and their potential failures. These are the Type B-tested components that have isolated but exhibit excessive leakage.
5	Independent (or random) isolation failures include those accidents in which the pre-existing isolation failure to seal is not dependent on the sequence in progress. This class is similar to Class 4 isolation failures, but is applicable to sequences involving Type C tests and their potential failures.
6	Containment isolation failures include those leak paths covered in the plant test and maintenance requirements or verified per in service inspection and testing (ISI/IST) program.
7	Accidents involving containment failure induced by severe accident phenomena. Changes in Appendix J testing requirements do not impact these accidents.
8	Accidents in which the containment is bypassed (either as an initial condition or induced by phenomena) are included in Class 8. Changes in Appendix J testing requirements do not impact these accidents.

Calvert Cliffs Liner Corrosion Analysis [5]

This submittal to the NRC describes a method for determining the change in likelihood, due to extending the ILRT, of detecting liner corrosion, and the corresponding change in risk. The methodology was developed for Calvert Cliffs in response to a request for additional information regarding how the potential leakage due to age-related degradation mechanisms was factored into the risk assessment for the ILRT one-time extension. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. The QCNPS containment is a pressure-suppression BWR/Mark I type with a steel shell in the drywell region, including a portion below the concrete drywell floor. The shell is surrounded by concrete.

EPRI 1018243 [3]

This report presents a risk impact assessment for extending ILRT surveillance intervals to 15 years. This risk impact assessment complements the earlier EPRI report, TR-104285 [2]. The earlier report considered changes to local leak rate testing intervals as well as changes to ILRT testing intervals. The original risk impact assessment [2] considers the change in risk based on population dose, whereas the revision [3] considers dose as well as large early release frequency (LERF) and conditional containment failure probability (CCFP). This report deals with changes to ILRT testing intervals and is intended to provide bases for supporting changes to industry and regulatory guidance on ILRT surveillance intervals.

The risk impact assessment using the Jeffrey's Non-Informative Prior statistical method is further supplemented with a sensitivity case using expert elicitation performed to address conservatisms. The expert elicitation is used to determine the relationship between pre-existing containment leakage probability and magnitude. The results of the expert elicitation process from this report are used as a separate sensitivity investigation for the QCNPS analysis presented here in Section 6.2.

NRC Safety Evaluation Report [7]

This SE documents the NRC staff's evaluation and acceptance of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, subject to the limitations and conditions identified in the SE and summarized in Section 4.0 of the SE. These limitations (associated with the ILRT Type A tests) were addressed in the Revision 2-A of NEI 94-01 which are also included in Revision 3-A of NEI 94-01 [1] and the final version of the updated EPRI report [3]. Additionally, the SE clearly defined the acceptance criteria to be used in future Type A ILRT extension risk assessments as delineated previously in the end of Section 1.3.

4.2 PLANT-SPECIFIC INPUTS

The QCNPS Unit 1 and Unit 2 specific information used to perform this ILRT interval extension risk assessment includes the following:

- Level 1 and Level 2 PRA model quantification results [16, 17]
- Population dose within a 50-mile radius for various release categories [8]

QCNPS Unit 1 and Unit 2 Internal Events Core Damage Frequencies

The current QCNPS Unit 1 and Unit 2 Internal Events PRA models of record are based on an event tree / linked fault tree model characteristic of the as-built, as-operated plant. Based on the results reported in References [16, 17], the internal events Level 1 PRA core damage frequency (CDF) is 2.91E-06/yr for QC1. Table 4.2-1 provides the CDF results by accident class.

No substantive differences exist between the QCNPS Unit 1 and Unit 2 that are judged to affect the conclusions of the PRA. As such, no separate PRA quantification is conducted for Unit 2. Since the QCNPS PRA Unit 1 PRA results are judged representative of both Unit 1 and Unit 2, the ILRT extension evaluation is considered applicable to both Unit 1 and Unit 2.

TABLE 4.2-1

QCNPS CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE SUBCLASS

ACCIDENT CLASS DESIGNATOR	SUBCLASS	DEFINITION	2014 MODEL (PER RX YR)
Class I	A(1)	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	1.08E-06
	B	Accident sequences involving a station blackout and loss of coolant inventory makeup. (Class IBE is defined as "Early" Station Blackout events with core damage at less than 4 hours. Class IBL is defined as "Late" Station Blackout events with core damage at greater than 4 hours.)	IBE (1.11E-07) IBL (2.86E-07)
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	2.11E-08
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.	4.63E-08
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage; core damage induced post containment failure.	9.82E-07
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage induced post containment failure. (Not used)	1.46E-09
	V	Classes IIA and III except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	5.19E-08
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	ε
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	8.17E-08
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	1.04E-07
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	1.97E-08
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	8.43E-08
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g. LOCA or SORV); core damage induced post containment failure.	
Class V	---	Unisolated LOCA outside containment.	4.70E-08
		Total	**2.91E-06**

Note to Table 4.2-1:

(1) Includes contribution from Class IE sequences.

QCNPS Internal Events Release Category Frequencies

The Level 2 Model that is used for QCNPS was developed to calculate the LERF contribution as well as the other release categories evaluated in the model. Table 4.2-1 summarizes the pertinent QCNPS Unit 1 results in terms of release category. The total Large Early Release Frequency (LERF) which corresponds to the H/E release category in Table 4.2-1 was found to be 1.97E-7/yr. The total release frequency is 2.51E-06/yr.

Table 4.2-2 provides release categories by accident class. Table 4.2.-3 provides release frequency contribution by accident class and by release category. Tables 4.2-2 and 4.2-3 results are from Table 3.4-4 of the PRA Summary Notebook [16]. Table 4.2-4 provides QC isolation failure sequence frequency contribution. The ILRT risk assessment methodology requires separate treatment of isolation failure sequences as these are addressed under EPRI Class 2. Information from Tables 4.2-2, 4.2-3 and 4.2-4 will be used in later sections for dose calculations.

TABLE 4.2-2

QCNPS LEVEL 2 PRA MODEL RELEASE CATEGORIES AND FREQUENCIES(1)

CATEGORY	FREQUENCY/YR(2)
Intact	3.98E-07
H/E – High Early (LERF)	1.97E-07
M/E – Medium Early	5.62E-08
L/E – Low Early	9.12E-09
LL/E – Low Low Early	0.00E+00
H/I – High Intermediate	1.29E-06
M/I – Medium Intermediate	8.99E-07
L/I – Low Intermediate	2.33E-11
LL/I – Low Low Intermediate	3.76E-08
H/L – High Late	1.72E-10
M/L – Medium Late	9.56E-09
L/L – Low Late	1.06E-08
LL/L – Low Low Late	5.72E-09
Total Release Frequency (Excluding Intact Frequency)	**2.51E-06**
Core Damage Frequency(1)	**2.92E-06**

Notes to Table 4.2-2:

(1) The Level 2 based accident class CDF total of 2.92E-6/yr is slightly higher than the Level 1 based CDF total of 2.91E-06/yr from the single top model due to the generation of non-minimal cutsets. The difference is minimal and does not impact the results.

(2) Table 4.2-2 data source is Table 3.4-4 from the PRA Summary Notebook [16]. Table 3.4-4 of the PRA Summary Notebook is reproduced as Table 4.2-3 on the next page of this risk assessment (see next page).

TABLE 4.2-3

SUMMARY OF QCNPS 2014A LEVEL 2 RELEASE CATEGORIES (/YR)[(1) (2)]

CLASS	BASE CDF	INTACT[(6)]	LL/E	LL/I	LL/L	L/E	L/I	L/L	M/E	M/I	M/L	H/E	H/I	H/L	TOTAL RELEASE
IA	1.08E-06	2.31E-07	0.00E+00	1.16E-08	8.74E-10	8.34E-09	0.00E+00	5.36E-10	1.43E-08	2.96E-07	6.63E-09	3.34E-08	4.76E-07	1.68E-10	8.48E-07
IBE	1.11E-07	1.35E-08	0.00E+00	1.84E-08	4.79E-11	4.19E-11	0.00E+00	3.20E-11	2.08E-11	1.53E-08	5.52E-10	8.91E-10	6.21E-08	2.22E-12	9.74E-08
IBL	2.86E-07	2.42E-08	0.00E+00	5.07E-09	4.71E-12	0.00E+00	2.33E-11	7.61E-12	0.00E+00	1.46E-08	2.35E-11	0.00E+00	2.42E-07	2.59E-12	2.62E-07
IC	2.11E-08	2.11E-08	0.00E+00	0.00E+00	0.00E+00	3.07E-11	0.00E+00	0.00E+00	0.00E+00	8.99E-12	0.00E+00	1.97E-12	0.00E+00	0.00E+00	4.17E-11
ID	4.63E-08	1.51E-08	0.00E+00	2.59E-09	1.14E-09	0.00E+00	0.00E+00	2.38E-09	0.00E+00	2.32E-09	6.16E-10	2.94E-09	1.92E-08	0.00E+00	3.12E-08
II	1.03E-06	ε	0.00E+00	3.38E-12	3.40E-12	0.00E+00	0.00E+00	6.97E-12	0.00E+00	5.22E-07	1.29E-12	4.30E-08	4.64E-07	0.00E+00	1.03E-06
IIIA	N/A	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IIIB	8.17E-08	8.03E-08	0.00E+00	0.00E+00	0.00E+00	7.04E-10	0.00E+00	0.00E+00	0.00E+00	2.88E-10	0.00E+00	3.88E-10	1.29E-11	0.00E+00	1.39E-09
IIIC	1.04E-07	1.24E-08	0.00E+00	0.00E+00	3.65E-09	0.00E+00	0.00E+00	7.64E-09	0.00E+00	4.87E-08	1.74E-09	7.47E-09	2.27E-08	0.00E+00	9.19E-08
IIID\	1.97E-08	ε	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.97E-08	0.00E+00	0.00E+00	1.97E-08
IV[(4)]	8.43E-08	ε	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.18E-08	8.99E-12	0.00E+00	4.25E-08	0.00E+00	0.00E+00	8.43E-08
V	4.70E-08	ε	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.70E-08	0.00E+00	0.00E+00	4.70E-08
Total	**2.92E-06[(5)]**	**3.98E-07**	**0.00E+00**	**3.76E-08**	**5.72E-09**	**9.12E-09**	**2.33E-11**	**1.06E-08**	**5.62E-08**	**8.99E-07**	**9.56E-09**	**1.97E-07**	**1.29E-06**	**1.72E-10**	**2.51E-06**

Notes to Table 4.2-3:

(1) Level 2 quantified at a truncation value of 1E-12/yr.

(2) N/A indicates that the accident class did not contribute to release of that specific category.

(3) Not used.

(4) The total Class IV release frequency was initially 8.45E-8/yr, which was slightly higher than the base Class IV CDF of 8.43E-8/yr due to success branch issues. The Class IV Moderate/Early (M/E) release frequency total was reduced by approximately 2E-10/yr to account for this minor discrepancy.

(5) The Accident Class CDF total of 2.92E-6/yr is slightly higher than the base CDF total of 2.91E-06/yr from the single top model due to the generation of non-minimal cutsets. The difference is minimal and does not impact the results.

(6) The intact frequency determined as the difference between base CDF and total release frequencies.

TABLE 4.2-4

QCNPS L2 ISOLATION FAILURE FREQUENCY CONTRIBUTION

2014A L2 SEQUENCE	CDF CONTRIBUTION(1)
RCVL2-3B-045	1.16E-10
RCVL2-3C-045	5.78E-09
RCVL2-IA-087	1.71E-09
RCVL2-IBE-087(2)	4.69E-10
RCVL2-IC-087	1.97E-12
RCVL2-ID-087	2.61E-09
Total Contribution	**1.07E-08**

Notes to Table 4.2-4:

(1) Source Table I-1 *Quad Cities PRA Level 2 Importance Measures (Sorted by F-V) (LERF = 1.97E-7/yr),* QCNPS Quantification Notebook [17]. These sequences are all categorized as high early (H/E) releases.

(2) RCVL2-IBL-087 is a containment isolation failure sequence that results in a high intermediate (H/I) release and is therefore not included in this table.

QCNPS Population Dose Information

The population dose is calculated by using data provided in Appendix F Severe Accident Mitigation Alternatives (SAMA) of the QCNPS License Renewal Report [8] and adjusting the results for the current QCNPS Unit 1 Level 2 model results, more recent population estimates, and current allowed technical specification leakage. Each accident class was associated with an applicable Accident Sequence from the SAMA evaluation. Table 4.2-5 reproduces the SAMA evaluation consequence categories L2-1 through L2-10 and the dominant release category for each consequence category.

The EPRI Class and SAMA Sequences are mapped in Table 4.2-6 to QCNPS Release Frequencies found in Table 4.2-3. The frequencies found in the bottom row of Table 4.2-3 are brought forward to Table 4.2-6. Class 7 sequences are accident progression bins in which containment failure induced by severe accident phenomena occurs. Each Class 7 endstate (7a, 7b, 7c, 7d or 7f) is mapped to a SAMA consequence category and each of the SAMA consequence releases. For example, EPRI Class 7f is mapped to SAMA consequence L2-8 and its releases of L/I, LL/I, L/L and LL/L.

TABLE 4.2-5

ACCIDENT SEQUENCE CATEGORY DESCRIPTIONS FROM THE LICENSE RENEWAL SAMA EVALUATION

CONSEQUENCE CATEGORY	DOMINANT RELEASE CATEGORY	MAAP CASE	TIME OF INITIAL RELEASE	TIME OF GEN. EMG. DECLARATION	TIME OF END OF RELEASE	EAL BASIS	RELEASE FREQUENCY (PER RX YR)
L2-1	H/E (LERF)	QC 0053 IA-L2-1A-NSPR	4.4 hr	60 min	36 hr	FG1	2.5E-7[(1)]
L2-2	H/I	QC 0082 IIA-L2-9C[(4)]	51.4 hr	15 hr	72 hr	HG2	4.1E-8[(2)]
L2-3	H/L	None	--	--	--	--	--
L2-4	M/E	QC-0085 IVA-L2-14B-ED-WW	55 min	55 min	36 hr	FG1	2.5E-7
L2-5	M/I	QC 0061 IIA-I2-9a	39.3 hr	15 hr	72 hr	HG2	8.0E-7[(3)]
L2-6	M/L	None	--	--	--	--	--
L2-7	L/E or LL/E	QC-057 ID-L2-7B NSPR	5.7 hr	45 min	36 hr	FG1	9.7E-9
L2-8	L/I or LL/I or L/L or LL/L	QC 0058 ID-L2-7BA-SPRY	25.9 hr	15 hr	36 hr	HG2	3.2E-7
L2-9	Class V	QC OO70 V-L2-17	17 min	20 min	36 hr	FG1	1.8E-8
L2-10	Intact	QC 0074 IB-L2-22	48 min	60 min	36 hr	FG1	5.0E-7

Notes to Table 4.2-5:

(1) Does not include Class V (see L2-9).

(2) Includes H/I and H/L.

(3) Includes M/I and M/L.

(4) Containment fails at 45.9 hr.

TABLE 4.2-6

RELEASE CATEGORIES MAPPED TO EPRI CLASS, SAMA SEQUENCES AND QC 2014A RELEASE FREQUENCY

EPRI CLASS	SAMA SEQ.	RELEASE CATEGORY(1)	SAMA DOSE (PERSON-REM)	QC 2014A FREQ.(2) (/YR)
--	--	Base CDF	--	2.92E-06
1	L2-10	Intact(6)	2.62E+03(7)	3.98E-07
7a	L2-1	H/E	2.16E+06	1.39E-07(4)
7b	L2-2	H/I	1.62E+06	1.29E-06
7b	L2-2(5)	H/L	1.62E+06	1.72E-10
7c	L2-4	M/E	1.53E+06	5.62E-08
7d	L2-5	M/I	6.14E+05	8.99E-07
7d	L2-5(3)	M/L	6.14E+05	9.56E-09
7e	L2-7	LL/E	8.54E+05	0.00E+00
7e	L2-7	L/E	8.54E+05	9.12E-09
7f	L2-8	LL/I	3.35E+05	3.76E-08
7f	L2-8	LL/L	3.35E+05	5.72E-09
7f	L2-8	L/I	3.35E+05	2.33E-11
7f	L2-8	L/L	3.35E+05	1.06E-08

Notes to Table 4.2-6:

(1) Release categories are consistent with release categories found in Table 4.2-4 and Table 4.2-3. These release categories are used to map dose (person-rem) from the SAMA evaluation to release categories frequencies from the 2014A Level 2 PRA model.

(2) Frequency is from Table 4.2.3 based on Release Category.

(3) No MAAP case was developed for M/L frequency. Therefore M/L is mapped to SAMA sequence L2-5 for a M/I release.

(4) Does not include frequency contribution from containment bypass (Class V) and H/E containment isolation failure sequences. Calculated as 1.97E-7/yr (total H/E) – 4.70E-8/yr (Class V) – 1.07E-8/yr (H/E cont isolation failure) = 1.39E-7/yr.

(5) No MAAP case was developed for H/L frequency. Therefore H/L is mapped to SAMA sequence L2-2 for a H/I release.

(6) The intact frequency determined as the difference between base CDF and total release frequencies (including Class V and isolation failure sequences).

(7) Quad Cities Technical Specification Basis Document B 3.6.1.1 “Primary Containment” [34] states, “The maximum allowable leakage rate for the primary containment (La) is 3.0% by weight of the containment air per 24 hours at a containment pressure of 43.9 psig.” The SAMA analysis used a maximum allowable leakage rate of 0.5%. Therefore, the SAMA dose of 4.36E+02 is increased by a factor of 6 for this ILRT analysis.

Population Estimate

The QCNPS SAMA dose results in Table 4.2-6 are based on a 50-mile population estimate developed from year 2000 census data and projected to year 2032. The total 50-mile population used in the SAMA analysis was 700,677, as documented in Table F-3 *Estimated Population Distribution Within a 50-Mile Radius of QCNPS, Year 2032* from Appendix F of the QC License Renewal Application.

Year 2010 census data is now readily available. As a reasonableness check, SecPop Version 4.2.0 [18] was used to calculate the 50-mile population around QCNPS using both the year 2000 and 2010 census data files within SecPop, yielding 650,094 persons for year 2000 and 648,591 persons for year 2010. Thus, SecPop shows a population decline for the 50-mile region surrounding QCNPS between year 2000-2010. Based on this population decline over that 10 year period, use of the SAMA projected value of 700,677 is judged reasonable.

Power Level and Containment Leak Rate

The parameters of reactor power level and containment leak rate that were used in the QCNPS SAMA analysis were also compared with the current QCNPS parameters to determine if both reflect the as-built, as-operated plant.

	POWER LEVEL	**CONTAINMENT LEAK RATE**
QC SAMA Analysis (2002)	2957 MWt(1)	0.5% wgt of cont. air /day(2)
QC 2016 ILRT Risk Assessment	2957 MWt(3)	3% wgt of cont. air /day(2)

Notes to Table:

(1) The 2002 SAMA Analysis used the Extended Power Uprate (EPU) Thermal Power of 2957 MWt. This is noted in Section F.3.5 of the SAMA Analysis which states, “The core inventory at the time of the accident was based on the input supplied in the MACCS User’s Guide (Reference 91). The core inventory corresponds to the end-of-cycle values for a 3412-MWth PWR plant. A scaling factor of 0.867 was used to provide a representative core inventory of 2957-MWth at QCNPS.”

(2) Containment Leak Rate Limit is found in TS Basis Document B 3.6.1.1 [33].

(3) The Reactor Power Level is found in Section 1.1 of the QCNPS Updated Safety Analysis Report [27].

Since the reactor power level and population are unchanged from the SAMA evaluation, no corrections are needed for these parameters. An adjustment is made to account for the change in leakage rate from 0.5% weight/day used in the 2002 SAMA analysis to the current L_a of 3% weight/day. A factor of 6 adjustment is applied to the Intact Containment release category SAMA dose (person-rem). The other SAMA release categories involve a failure of containment such that an adjustment for containment leakage rate is not required. This approach provides a first-order approximation for QCNPS of the population doses associated with each of the release categories from the SAMA evaluation. This is considered adequate since the conclusions from this analysis will not be substantially affected by the actual dose values that are used.

4.3 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 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 accounted for, the EPRI Class 3 accident class as defined in Table 4.1-1 is divided into two sub-classes representing small and large leakage failures. These subclasses are defined as Class 3a and Class 3b, respectively.

The probability of the EPRI Class 3a failures may be determined, consistent with the latest EPRI guidance [3], as the mean failure estimated from the available data (i.e., 2 “small” failures that could only have been discovered by the ILRT in 217 tests leads to a 2/217=0.0092 mean value). For Class 3b, consistent with latest available EPRI data, a non-informative prior distribution is assumed for no “large” failures in 217 tests (i.e., 0.5/(217+1) = 0.0023).

The EPRI methodology contains 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. This information includes a discussion of conservatisms in the quantitative guidance for delta LERF. EPRI describes ways to demonstrate that, using plant-specific calculations, the delta LERF is smaller than that calculated by the simplified method.

The methodology 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 QCNPS (as detailed in Section 5) means that the Class 2, Class 7, and Class 8 LERF sequences are subtracted from the CDF that is applied to Class 3b. To be consistent, the same change is made to the Class 3a CDF, even though these events are not considered LERF. Note that Class 2 events refer to sequences with a large pre-existing containment isolation failure that lead to LERF, a subset of Class 7 events are LERF sequences due to an early containment failure from energetic phenomena, and Class 8 are containment bypass events that contribute to LERF.

Consistent with the EPRI methodology [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 yr / 2), and the average time that a leak could exist without detection for a ten-year interval is 5 years (10 yr / 2). This ILRT interval 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, given a 10-year vs. a 3-yr interval. Correspondingly, an extension of the ILRT interval to fifteen years can be estimated to lead to about a factor of 5.0 (7.5/1.5) increase in the non-detection probability of a leak.

QCNPS Unit 1 and Unit 2 Past ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 under option B criteria is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart) where the calculated performance leakage rate was less than 1.0 La, and in compliance with the performance factors in NEI 94-01, Section 11.3. Based on the successful completion of two consecutive ILRTs at QCNPS Unit 1 and Unit 2, the current ILRT interval is once per ten years [31]. Note that the probability of a pre-existing leakage due to extending the ILRT interval is based on the industry-wide historical results as noted in the EPRI guidance document [3].

EPRI Methodology

This analysis uses the approach outlined in the EPRI Methodology [3]. The six steps of the methodology are:

1. Quantify the baseline (three-year ILRT frequency) risk in terms of frequency per reactor year for the EPRI accident classes of interest.
2. Develop the baseline population dose (person-rem, from the plant PRA or IPE, or calculated based on leakage) for the applicable accident classes.
3. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.
4. Determine the risk impact in terms of the change in LERF.
5. Determine the impact on the Conditional Containment Failure Probability (CCFP).
6. Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, and external event impacts.

The first three steps of the methodology deal with calculating the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The fourth step in the methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because the change in CDF for QCNPS is minimal, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The fifth step of the methodology calculates the change in containment failure probability, referred to as the conditional containment failure probability (CCFP). The NRC has identified a CCFP of less than 1.5% as the acceptance criteria for extending the Type A ILRT test intervals as the basis for showing that the proposed change is consistent with the defense in depth philosophy [7]. As such, this step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174. Step 6 takes into consideration the additional risk due to external events, and investigates the impact on results due to varying the assumptions associated with the liner corrosion rate and failure to visually identify pre-existing flaws.

4.4 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 the methodology from the Calvert Cliffs liner corrosion analysis [5]. The Calvert Cliffs analysis was performed for a concrete cylinder and dome and a concrete basemat, each with a steel liner. The QCNPS containment is a pressure-suppression BWR/Mark I type with a steel shell in the drywell region, including a portion below the concrete drywell floor. The shell is surrounded by concrete.

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 head
- 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>

1. Based on a review of industry events, an Oyster Creek incident is assumed to be applicable to QCNPS for a concealed shell failure in the floor. In the Calvert Cliffs analysis, this event was assumed not to be applicable and 0.5 failures were assumed (i.e. a typical PRA model when no failures have been identified). For QCNPS one failure (rather than 0.5) is assumed for the floor area. (See Table 4.4-1, Step 1, Containment Basement probability calculation.)
2. The two corrosion events over a 5.5 year data period are used to estimate the liner flaw probability in the Calvert Cliffs analysis and are assumed to be applicable to the QCNPS containment analysis. These events, one at North Anna Unit 2 and one at Brunswick Unit 2, were initiated from the non-visible (backside) portion of the containment liner. It is noted that two additional events have occurred in recent years (based on a data search covering approximately 9 years documented in Reference [30]). In November 2006, the Turkey Point 4 containment building liner developed a hole when a sump pump support plate was moved. In May 2009, a hole approximately 3/8" by 1" in size was identified in the Beaver Valley 1 containment liner. For risk evaluation purposes, these two more recent events occurring over a 9 year period are judged to be adequately represented by the two events in the 5.5 year period of the Calvert Cliffs analysis incorporated in the EPRI guidance (See Table 4.4-1, Step 1).
3. 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 4.4-1, Steps 2 and 3). Sensitivity studies are included that address doubling this rate every two years and every ten years.
4. 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 region, and 0.11% (10% of the cylinder failure probability) for the basemat. These values were determined from an assessment of the containment fragility curve versus the ILRT test pressure. For QCNPS the containment failure probabilities are conservatively assumed to be 10% for the drywell outer walls and 1% for the basemat. Sensitivity studies are included that increase and decrease the probabilities by an order of magnitude. (See Table 4.4-1, Step 4.)
5. 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 for the containment cylinder and head. For the containment basemat, 100% is assumed unavailable for visual inspection. To date, all liner corrosion events have been detected through visual inspection (See Table 4.4-1, Step 5). The Calvert Cliffs analysis is based on an estimate of 85% of the interior wall surface being visible for inspection. QCNPS estimates that at least 94% of the interior surface of the QCNPS containment wall is inspectable [33]. Therefore, use of the Calvert Cliff's analysis is conservative. Sensitivity studies are included that evaluate total detection failure likelihood of 5% and 15%, respectively.

6. 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 4.4-1

STEEL LINER CORROSION BASE CASE

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND HEAD		CONTAINMENT BASEMAT	
1	**Historical Steel Liner Flaw Likelihood** Failure Data: Containment location specific (consistent with Calvert Cliffs analysis).	Events: 2 2/(70 * 5.5) = **5.2E-3**		Events: 1 1.0/(70 * 5.5) = **2.6E-3**	
2	**Age Adjusted Steel Liner Flaw Likelihood** During 15-year interval, assume failure rate doubles every five years (14.9% increase per year). The average for 5th to 10th year is set to the historical failure rate (consistent with Calvert Cliffs analysis).	<u>Year</u> 1 avg 5-10 15	<u>Failure Rate</u> 2.1E-3 5.2E-3 1.4E-2	<u>Year</u> 1 avg 5-10 15	<u>Failure Rate</u> 1.0E-3 2.6E-3 7.0E-3
		15 year average = 6.27E-3		**15 year average = 3.14E-3**	
3	**Flaw Likelihood at 3, 10, and 15 years** Uses age adjusted liner flaw likelihood (Step 2), assuming failure rate doubles every five years (consistent with Calvert Cliffs analysis – See Table 6 of Reference [5]).	**0.71% (1 to 3 years)** **4.06% (1 to 10 years)** **9.40% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 8.7% to utilize in the estimation of the delta-LERF value. For this analysis, the values are calculated based on the 3, 10, and 15 year intervals.)		**0.36% (1 to 3 years)** **2.03% (1 to 10 years)** **4.70% (1 to 15 years)** (Note that the Calvert Cliffs analysis presents the delta between 3 and 15 years of 2.2% to utilize in the estimation of the delta-LERF value. For this analysis, twice that value is utilized (since 1 failure is assumed applicable instead of 0.5) and the values are calculated based on the 3, 10, and 15 year intervals.)	
4	**Likelihood of Breach in Containment Given Steel Liner Flaw** The failure probability of the containment cylinder and dome is assumed to be 10% (compared to 1.1% in the Calvert Cliffs analysis). The basemat failure probability is assumed to be a factor of ten less, 1% (compared to 0.11% in the Calvert Cliffs analysis).	**10%**		**1%**	

TABLE 4.4-1

STEEL LINER CORROSION BASE CASE

STEP	DESCRIPTION	CONTAINMENT CYLINDER AND HEAD	CONTAINMENT BASEMAT
5	**Visual Inspection Detection Failure Likelihood** Utilize assumptions consistent with Calvert Cliffs analysis.	**10%** 5% failure to identify visual flaws plus 5% likelihood that the flaw is not visible (not through-cylinder but could be detected by ILRT) All events have been detected through visual inspection. 5% visible failure detection is a conservative assumption.	**100%** Cannot be visually inspected.
6	**Likelihood of Non-Detected Containment Leakage** (Steps 3 * 4 * 5)	**0.0071% (at 3 years)** =0.71% * 10% * 10% **0.0406% (at 10 years)** =4.06% * 10% * 10% **0.0940% (at 15 years)** =9.40% * 10% * 10%	**0.0036% (at 3 years)** =0.36% * 1% * 100% **0.0203% (at 10 years)** =2.03% * 1% * 100% **0.0470% (at 15 years)** =4.70% * 1% * 100%

The total likelihood of the corrosion-induced, non-detected containment leakage that is subsequently added to the EPRI Class 3b contribution is the sum of Step 6 for the containment cylinder and dome, and the containment basemat:

- At 3 years: 0.0071% + 0.0036% = 0.0107%
- At 10 years: 0.0406% + 0.0203% = 0.0609%
- At 15 years: 0.0940% + 0.0470% = 0.1410%

5.0 RESULTS

The application of the approach based on EPRI Guidance [3] has led to the following results. The results are displayed according to the eight accident classes defined in the EPRI report. Table 5.0-1 lists these accident classes.

TABLE 5.0-1

ACCIDENT CLASSES

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION
1	Containment Intact
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 analysis performed examined the QCNPS specific accident sequences in which the containment remains intact or the containment is impaired. Specifically, the categorization of the severe accidents contributing to risk was considered in the following manner:

- Core damage sequences in which the containment remains intact initially and in the long term (EPRI Class 1 sequences).
- Core damage sequences in which containment integrity is impaired due to random isolation failures of plant components other than those associated with Type B or Type C test components. For example, liner breach or bellows leakage, if applicable. (EPRI Class 3 sequences).

- Core damage sequences in which containment integrity is impaired due to containment isolation failures of pathways left "opened" following a plant post-maintenance test. (For example, a valve failing to close following a valve stroke test. (EPRI Class 6 sequences). Consistent with the EPRI Guidance, this class is not specifically examined since it will not significantly influence the results of this analysis.
- Accident sequences involving containment bypass (EPRI Class 8 sequences), large containment isolation failures (EPRI Class 2 sequences), and small containment isolation "failure-to-seal" events (EPRI Class 4 and 5 sequences) are accounted for in this evaluation as part of the baseline risk profile. However, they are not affected by the 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.

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

Step 1 Quantify the base-line risk in terms of frequency per reactor year for each of the accident classes presented in Table 5.0-1.

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

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

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

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

Step 6 Evaluate the sensitivity of the results to assumptions in the liner corrosion analysis, and external event impacts.

It is noted that the calculations were generally performed using an electronic spreadsheet such that the presented numerical results may differ slightly as compared to values calculated by hand.

5.1 STEP 1 – QUANTIFY THE BASE-LINE RISK IN TERMS OF FREQUENCY PER REACTOR YEAR

This step involves the review of the QCNPS Level 2 accident sequence frequency results. Table 5.1-1 relates EPRI class containment release scenarios to accident sequence categories used in the SAMA evaluation for the QC license renewal application. This application combined with the QCNPS dose (person-rem) results mapping documented in Table 4.2-6 forms the basis for estimating the population dose for QCNPS.

For the assessment of ILRT impacts on the risk profile, the potential for pre-existing leaks is included in the model. (These events are represented by the Class 3 sequences in EPRI TR-1018243 [3]). 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 5.0-1 were developed for QCNPS based on Level 2 PRA inputs found in Section 4, determining the frequencies for Classes 3a and 3b, and then determining the remaining frequency for Class 1. Furthermore, adjustments were made to the Class 3b and hence Class 1 frequencies to account for the impact of undetected corrosion of the steel liner per the methodology described in Section 4.4. The eight containment release class frequencies were developed consistent with the definitions in Table 5.0-1 as described following Table 5.1-1.

Table 5.1-1 provides dose values for each EPRI scenario class. The dose values were developed in Section 4.2. The Level 2 Accident sequence bin(s) assigned to each EPRI Class are described under each containment release class discussion following Table 5.1-1. The methodology for determining the dose applied to EPRI Class 7 is further described under the paragraph heading “Class 7 Sequences”.

TABLE 5.1-1

EPRI CLASS DOSE ASSIGNMENT FROM THE QC SAMA CONSEQUENCE MODEL

EPRI SCENARIO CLASS	LEVEL 2 ACCIDENT SEQUENCE BIN	POPULATION DOSE (PERSON-REM)[3]
1	L2-10 (Containment Intact)	2.62E+03
2	L2-1[1] (Isolation Failure)	2.16E+06
7	All EPRI Class 7a through 7f Level 2 bins	1.25E+06[2]
7a (H/E)	L2-1 (H/E)	2.16E+06
7b (H/I or H/L)	L2-2 (H/I)	1.62E+06
7c (M/E)	L2-4 (M/E)	1.53E+06
7d (M/I or M/L)	L2-5 (M/I)	6.14E+05
7e (L/E or LL/E)	L2-7 (L/E or LL/E)	8.54E+05
7f (L/I, LL/I, L/L, or LL/L)	L2-8 (L/I, LL/I, L/L, or LL/L)	3.35E+05
8	L2-9 (Containment Bypass)	4.11E+06

Notes to Table 5.1-1:

(1) QC SAMA sequence L2-1 represents the highest containment failure (non-containment bypass) dose.

(2) Given that multiple QCNPS discrete scenarios apply to the broader EPRI Class 7, the EPRI dose is based on a weighted average based on QCNPS 2014A scenario frequencies). The weighted average dose is developed in Table 5.1-2.

(3) Values are from the SAMA dose analysis for the EPRI category as discussed in Section 4.2 and Table 4.2-6. No adjustments were required for population or power level. A factor of 6 increase was applied to the intact containment release based on an increase in L_a containment leakage rate of 0.5% wgt/day for SAMA analysis to the current TS [34] allowable rate of 3.0% wgt/day (6 * 4.26E+02 person-rem = 2.62E+03 person-rem).

Class 1 Sequences

This group consists of all core damage accident progression bins for which the containment remains intact (modeled as Technical Specification Leakage). The frequency per year for these sequences is 3.76E-07/yr and is determined by subtracting all containment failure end states including the EPRI/NEI Class 3a and 3b frequency calculated below, from the total CDF.

Class 1 = CDF – (EPRI Classes)
= 2.92E-06 – (1.07E-08 (class 2) + 2.63E-08 (3a) + 6.58E-09 (3b) + 1.39E-07 (7 LERF) + 2.31E-06 (7-Non-LERF) + 4.70E-08 (Class 8))
= 3.76E-07/yr (from Excel^{TM} spreadsheet calculation)

For this analysis, the associated maximum containment leakage for this group is 1 La, consistent with an intact containment evaluation.

Class 2 Sequences

This group consists of containment isolation failures. For QCNPS, all containment isolation failure sequences result in a large early release and were assigned to accident sequence bin L2-1 H/E (LERF). The QC L2 Sequences associated with containment isolation failure leading to a large early release are the following: IA-087, IBE-087, IC-087, ID-087, 3B-045, and 3C-045. The sum of the frequencies of these scenarios is 1.07E-08/yr. For QCNPS one containment isolation failure sequence (IBL-087) was not included because this sequence is assessed as a non-early release. Sequence IBL-087 is included in the Class 7 sequences categorized as L2-2 H/I accident sequences. Note that Class 2 frequency is not affected by the ILRT interval change.

Class 3 Sequences

This group represents pre-existing leakage in the containment structure (e.g., containment liner). The containment leakage for these sequences can be either small (in excess of design allowable but <10La) or large. In this analysis, a value of 10 La was used for small pre-existing flaws and 100 La for relatively large flaws.

The respective frequencies per year are determined as follows:

$PROB_{Class\_3a}$ = probability of small pre-existing containment liner leakage
= 0.0092 (see Section 4.3)

$PROB_{Class\_3b}$ = probability of large pre-existing containment liner leakage
= 0.0023 (see Section 4.3)

As described in Section 4.3, additional consideration is made to not apply these failure probabilities to those cases that are already considered LERF scenarios (i.e., the Class 2 and Class 8 contributions). Note that some portion of the EPRI Class 7 frequency also represents LERF scenarios, but these are conservatively not subtracted from that portion of CDF eligible for EPRI Class 3. Additionally, CDF associated with failures that would never lead to LERF (e.g., Class II and Class IBL sequences) could also be excluded from EPRI Class 3a, but this is conservatively not performed. The adjustment to exclude EPRI Class 2 and Class 8 is made on the frequency information as shown below.

Class_3a = 0.0092 * [CDF – (Class 2 + Class 8)]
= 0.0092 * [2.92E-06 – (1.07E-08 + 4.70E-08)]
= 2.63E-08/yr

Class_3b = 0.0023 * [CDF – (Class 2 + Class 8)]
= 0.0023 * [2.92E-06 – (1.07E-08 + 4.70E-08)]
= 6.58E-09/yr

For this analysis, the associated containment leakage for Class 3a is 10 La and 100 La for Class 3b, which is consistent with the latest EPRI methodology [3] and the NRC SE [7].

Class 4 Sequences

This group represents containment isolation failure-to-seal of Type B test components. 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 this analysis.

Class 5 Sequences

This group represents containment isolation failure-to-seal of Type C test components. Because these 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.

Class 6 Sequences

This group is similar to Class 2. These are sequences that involve core damage with a failure-to-seal containment leakage due to failure to isolate the containment. These sequences are dominated by misalignment of containment isolation valves following a test/maintenance evolution. Consistent with the EPRI guidance [3], this accident class is not explicitly considered since it has a negligible impact on the results.

Class 7 Sequences

This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs. Note that containment failure is not induced for containment bypass (BOC and ISLOCA) (EPRI Class 8) and isolation failure (EPRI Class 2) sequences as these are either the initiating event or a plant condition, existing at the time of the initiating event. For this analysis, the associated radionuclide releases are based on the application of the Level 2 end states from the QCNPS SAMA evaluation as described in Section 4.2. The Class 7 Sequences are all Level 2 Sequences except containment intact EPRI Class 1, the containment bypass (EPRI Class 8) and isolation failure (EPRI Class 2) sequences leading to a large early release. The failure frequency and population dose for each specific release category is shown below in Table 5.1-2. The total release frequency and total dose are then used to determine a weighted average person-rem. The resulting weighted average person-rem is the representative EPRI Class 7 dose in the subsequent analysis. Note that the total frequency and dose associated from this EPRI class does not change as part of the ILRT extension request.

TABLE 5.1-2

ACCIDENT CLASS 7 FAILURE FREQUENCIES AND POPULATION DOSES (QC BASE CASE LEVEL 2 MODEL)

ACCIDENT CLASS	SAMA RELEASE CATEGORY	2014 PRA RELEASE FREQUENCY / YR	POPULATION DOSE (50 MILES) PERSON-REM (1)	POPULATION DOSE RISK (50 MILES) (PERSON-REM / YR)(2)
EPRI #7a (H/E)	L2-1	1.39E-07	2.16E+06	3.01E-01
EPRI #7b (H/I or H/L)	L2-2	1.29E-06	1.62E+06	2.08E+00
EPRI #7c (M/E)	L2-4	5.62E-08	1.53E+06	8.61E-02
EPRI #7d (M/I or M/L)	L2-5	9.09E-07	6.14E+05	5.58E-01
EPRI #7e (L/E or LL/E)	L2-7	9.12E-09	8.54E+05	7.79E-03
EPRI #7f (L/I, LL/I, L/L, or LL/L)	L2-8	5.39E-08	3.35E+05	1.81E-02
Class 7 Total	**--**	**2.45E-06(4)**	**1.25E+06(3)**	**3.06E+00**

Notes to Table 5.1-2:

(1) Population dose values from Table 5.1-1.

(2) Obtained by multiplying the Release Frequency per year by the Population Dose Person-Rem value. Calculations were performed using more than 3 significant figures. Therefore, figures may differ in the 3rd digit if one multiples the figures shown above.

(3) The weighted average population dose for Class 7 is obtained by dividing the total population dose risk by the total release frequency.

(4) Totals are from EXCEL spreadsheet using more than 3 significant figures.

Class 8 Sequences

This group represents sequences where containment bypass occurs (BOC, ISLOCA). For QCNPS, all containment bypass sequences were assigned L2-9 dose results. The sum of the frequencies of these scenarios is 4.70E-08/yr.

Summary of Accident Class Frequencies

In summary, the accident sequence frequencies that can lead to release of radionuclides to the public have been derived in a manner consistent with the definition of accident classes defined in EPRI 1018243 [3] and are shown in Table 5.1-3 by accident class.

TABLE 5.1-3

RADIONUCLIDE RELEASE FREQUENCIES AS A FUNCTION OF ACCIDENT CLASS (QCNPS BASE CASE)

ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	DESCRIPTION	FREQUENCY (PER RX-YR)
1	No Containment Failure	3.76E-07
2	Large Isolation Failures (Failure to Close)	1.07E-08
3a	Small Isolation Failures (liner breach)	2.63E-08
3b	Large Isolation Failures (liner breach)	6.58E-09
4	Small Isolation Failures (Failure to seal –Type B)	N/A
5	Small Isolation Failures (Failure to seal—Type C)	N/A
6	Other Isolation Failures (e.g., dependent failures)	N/A
7	Failures Induced by Phenomena (Early and Late)	2.45E-06
8	Bypass (Interfacing System LOCA)	4.70E-08
CDF	**All CET End states (including very low and no release)**	**2.92E-06**

5.2 STEP 2 – DEVELOP PLANT-SPECIFIC PERSON-REM DOSE (POPULATION DOSE) PER REACTOR YEAR

Plant-specific release analyses were performed to estimate the person-rem doses to the population within a 50-mile radius from the plant. The releases are based on information provided by Appendix F of the Severe Accident Mitigation Alternatives (SAMA) analysis of the QCNPS License Renewal Report [8]. The results of applying these releases to the EPRI/NEI containment failure classification are as follows:

Class 1	=	2.62E+03 person-rem (at 1.0 La)(1)	
Class 2	=	2.16E+06 person-rem(2)	
Class 3a	=	2.62E+03 person-rem x 10 La =	2.62E+04 person-rem(3)
Class 3b	=	2.62E+03 person-rem x 100 La =	2.62E+05 person-rem(3)
Class 4	=	Not analyzed	
Class 5	=	Not analyzed	
Class 6	=	Not analyzed	
Class 7	=	1.25E+06 person-rem(4)	
Class 8	=	4.11E+06 person-rem(5)	

In summary, the population dose estimates derived for use in the risk evaluation per the EPRI methodology [3] containment failure classifications are provided in Table 5.2-1.

(1) The Class 1, containment intact sequences, dose is assigned from the QC SAMA sequence L2-10 (Containment Intact) from the SAMA Level 3 adjusted dose for Quad Cities as shown in Table 5.1-1.

(2) The Class 2, containment isolation failures, dose is approximated from QC SAMA sequence L2-1 (Isolation Failure) from Table 5.1-1.

(3) The Class 3a and 3b dose are related to the leakage rate as shown, based on the EPRI methodology.

(4) The Class 7 dose is assigned from the weighted average dose calculated from QC SAMA sequence bins L2-1, L2-2, L2-4, L2-5, L2-7, L2-7, and L2-8 from Table 5.1-1 as detailed in Table 5.1-2 above.

(5) Class 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this class are assigned from QC SAMA sequence bin L2-9 from Table 5.1-1.

TABLE 5.2-1

QCNPS POPULATION DOSE ESTIMATES FOR POPULATION WITHIN 50 MILES

EPRI ACCIDENT CLASSES (CONTAINMENT RELEASE TYPE)	REPRESENTATIVE ACCIDENT PROGRESSION	DESCRIPTION	PERSON-REM (50 MILES)
1	Containment Intact	No Containment Failure (1 La)	2.62E+03
2	H/E (isolation failure; non-BOC, non-ISLOCA)	Large Isolation Failures (Failure to Close)	2.16E+06
3a	10 La	Small Isolation Failures (liner breach)	2.62E+04
3b	100 La	Large Isolation Failures (liner breach)	2.62E+05
4	N/A	Small Isolation Failures (Failure to seal –Type B)	NA
5	N/A	Small Isolation Failures (Failure to seal—Type C)	NA
6	N/A	Other Isolation Failures (e.g., dependent failures)	NA
7	See Table 5.1-2 (All releases except isolation, and bypass sequences)	Failures Induced by Phenomena (Early and Late)	1.25E+06
8	H/E (BOC, ISLOCA)	Bypass (Break Outside Containment or Interfacing System LOCA)	4.11E+06

The above dose estimates, when combined with the frequency results presented in Table 5.1-3, yield the QCNPS baseline mean consequence measures for each accident class. These results are presented in Table 5.2-2.

TABLE 5.2-2

QCNPS ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 3 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONT. RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR)(1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact (2)	2.62E+03	3.76E-07	9.82E-04	3.75E-07	9.82E-04	-7.97E-07
2	Large Isolation Failures (Failure to Close)	2.16E+06	1.07E-08	2.31E-02	1.07E-08	2.31E-02	--
3a	Small Isolation Failures (liner breach)	2.62E+04	2.63E-08	6.89E-04	2.63E-08	6.89E-04	--
3b	Large Isolation Failures (liner breach)	2.62E+05	6.58E-09	1.72E-03	6.89E-09	1.80E-03	7.97E-05
7	Failures Induced by Phenomena (Early and Late)	1.25E+06	2.45E-06	3.06E+00	2.45E-06	3.06E+00	--
8	Containment Bypass (Interfacing System LOCA)	4.11E+06	4.70E-08	1.93E-01	4.70E-08	1.93E-01	--
CDF	**All CET end states**		**2.92E-06**	**3.275**	**2.92E-06**	**3.275**	**7.89E-05**

Notes to Table 5.2-2:

(1) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

(2) Characterized as 1 La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.3 STEP 3 – EVALUATE RISK IMPACT OF EXTENDING TYPE A TEST INTERVAL FROM 10-TO-15 YEARS

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

Risk Impact Due to 10-year Test Interval

As previously stated, Type A tests impact only Class 3 sequences. For Class 3 sequences, the release magnitude is not impacted by the change in test interval (a small or large breach remains the same, even though the probability of not detecting the breach increases). Thus, only the frequency of Class 3a and 3b sequences is impacted. The risk contribution is changed based on the EPRI guidance as described in Section 4.3 by a factor of 3.33 compared to the base case values. The results of the calculation for a 10-year interval are presented in Table 5.3-1.

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 not detecting a leak in Classes 3a and 3b. For this case, the value used in the analysis is a factor of 5.0 compared to the 3-year interval value, as described in Section 4.3. The results for this calculation are presented in Table 5.3-2.

TABLE 5.3-1

QCNPS ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 10 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONT. RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) [1]
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact [2]	2.62E+03	2.99E-07	7.82E-04	2.97E-07	7.77E-04	-4.56E-06
2	Large Isolation Failures (Failure to Close)	2.16E+06	1.07E-08	2.31E-02	1.07E-08	2.31E-02	--
3a	Small Isolation Failures (liner breach)	2.62E+04	8.77E-08	2.29E-03	8.77E-08	2.29E-03	--
3b	Large Isolation Failures (liner breach)	2.62E+05	2.19E-08	5.73E-03	2.37E-08	6.19E-03	4.56E-04
7	Failures Induced by Phenomena (Early and Late)	1.25E+06	2.45E-06	3.06E+00	2.45E-06	3.06E+00	--
8	Containment Bypass (Interfacing System LOCA)	4.11E+06	4.70E-08	1.93E-01	4.70E-08	1.93E-01	--
CDF	**All CET end states**		**2.92E-06**	**3.280**	**2.92E-06**	**3.281**	**4.51E-04**

Notes to Table 5.3-1:

(1) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

(2) Characterized as 1 La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

TABLE 5.3-2

QCNPS ANNUAL DOSE AS A FUNCTION OF ACCIDENT CLASS; CHARACTERISTIC OF CONDITIONS FOR 1 IN 15 YEAR ILRT FREQUENCY

ACCIDENT CLASSES (CONT. RELEASE TYPE)	DESCRIPTION	PERSON-REM (0-50 MILES)	EPRI METHODOLOGY		EPRI METHODOLOGY PLUS CORROSION		CHANGE DUE TO CORROSION (PERSON-REM/YR) (1)
			FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	FREQUENCY (1/YR)	PERSON-REM/YR (0-50 MILES)	
1	Containment Intact (2)	2.62E+03	2.44E-07	6.38E-04	2.40E-07	6.27E-04	-1.06E-05
2	Large Isolation Failures (Failure to Close)	2.16E+06	1.07E-08	2.31E-02	1.07E-08	2.31E-02	--
3a	Small Isolation Failures (liner breach)	2.62E+04	1.32E-07	3.44E-03	1.32E-07	3.44E-03	--
3b	Large Isolation Failures (liner breach)	2.62E+05	3.29E-08	8.61E-03	3.70E-08	9.67E-03	1.06E-03
7	Failures Induced by Phenomena (Early and Late)	1.25E+06	2.45E-06	3.06E+00	2.45E-06	3.06E+00	--
8	Containment Bypass (Interfacing System LOCA)	4.11E+06	4.70E-08	1.93E-01	4.70E-08	1.93E-01	--
CDF	**All CET end states**		**2.92E-06**	**3.284**	**2.92E-06**	**3.285**	**1.05E-03**

Notes to Table 5.3-2:

(1) Only release Classes 1 and 3b are affected by the corrosion analysis. During the 15-year interval, the failure rate is assumed to double every five years. The additional frequency added to Class 3b is subtracted from Class 1 and the population dose rates are recalculated. This results in a small reduction to the Class 1 dose rate and an increase to the Class 3b dose rate.

(2) Characterized as 1 La release magnitude consistent with the derivation of the ILRT non-detection failure probability for ILRTs. Release Classes 3a and 3b include failures of containment to meet the Technical Specification leak rate.

5.4 STEP 4 – DETERMINE THE CHANGE IN RISK IN TERMS OF LARGE EARLY RELEASE FREQUENCY

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines "very small" changes in risk as resulting in increases of core damage frequency (CDF) below 1E-06/yr and increases in LERF below 1E-07/yr, and "small" changes in LERF as below 1E-06/yr. Because the ILRT for QCNPS has only a minor impact on CDF, the relevant metric is LERF.

For QCNPS, 100% of the frequency of Class 3b sequences can be used as a conservative first-order estimate to approximate the potential increase in LERF from the ILRT interval extension (consistent with the EPRI guidance methodology and the NRC SE). Based on the original 3-in-10 year test interval assessment from Table 5.2-2, the Class 3b frequency is 6.89E-09/yr, which includes the corrosion effect of the containment liner. Based on a ten-year test interval from Table 5.3-1, the Class 3b frequency is 2.37E-08/yr; and, based on a fifteen-year test interval from Table 5.3-2, it is 3.70E-08/yr. 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 (including corrosion effects) is 3.01E-08/yr. Similarly, the increase in LERF due to increasing the interval from 10 to 15 years (including corrosion effects) is 1.33E-08/yr. As can be seen, even with the conservatisms included in the evaluation (per the EPRI methodology), the estimated change in LERF is well within Region III of Figure 4 of Reference [4] (i.e., the acceptance criteria for "very small" changes in LERF) when comparing the 15 year results to the original 3-in-10 year requirement.

5.5 STEP 5 – DETERMINE THE IMPACT ON THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY

Another parameter that can provide input into the decision-making process is the change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The CCFP can be calculated from the results of this analysis. One of the difficult aspects of this calculation is providing a definition of the "failed containment." In this assessment, the CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state and, consistent with the EPRI guidance, the small isolation failures (Class 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

The change in CCFP can be calculated by using the method specified in the EPRI methodology [3]. The NRC SE has noted a change in CCFP of <1.5% as the acceptance criterion to be used as the basis for showing that the proposed change is consistent with the defense-in-depth philosophy. Table 5.5-1 shows the CCFP values that result from the assessment for the various testing intervals including corrosion effects in which the flaw rate is assumed to double every five years. The CCFP is calculated as follows:

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

TABLE 5.5-1

QCNPS ILRT CONDITIONAL CONTAINMENT FAILURE PROBABILITIES

CCFP 3 IN 10 YRS	**CCFP 1 IN 10 YRS**	**CCFP 1 IN 15 YRS**	**ΔCCFP$_{15\text{-}3}$**	**ΔCCFP$_{15\text{-}10}$**
86.25%	86.82%	87.28%	1.03%	0.46%

The change in CCFP of about 1% as a result of extending the test interval to 15 years from the original 3-in-10 year requirement is judged to be relatively insignificant, and is less than the NRC SE acceptance criteria of < 1.5%.

5.6 SUMMARY OF INTERNAL EVENTS RESULTS

Table 5.6-1 summarizes the internal events results of this ILRT extension risk assessment for QCNPS. The results between the 3-in-10 year interval and the 15 year interval compared to the acceptance criteria are then shown in Table 5.6-2, and it is demonstrated that the acceptance criteria are met.

TABLE 5.6-1

QCNPS ILRT CASES:
BASE, 3 TO 10, AND 3 TO 15 YR EXTENSIONS
(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	2.62E+03	3.75E-07	9.82E-04	2.97E-07	7.77E-04	2.40E-07	6.27E-04
2	2.16E+06	1.07E-08	2.31E-02	1.07E-08	2.31E-02	1.07E-08	2.31E-02
3a	2.62E+04	2.63E-08	6.89E-04	8.77E-08	2.29E-03	1.32E-07	3.44E-03
3b	2.62E+05	6.89E-09	1.80E-03	2.37E-08	6.19E-03	3.70E-08	9.67E-03
7	1.25E+06	2.45E-06	3.06E+00	2.45E-06	3.06E+00	2.45E-06	3.06E+00
8	4.11E+06	4.70E-08	1.93E-01	4.70E-08	1.93E-01	4.70E-08	1.93E-01
Total		**2.92E-06**	**3.275**	**2.92E-06**	**3.281**	**2.92E-06**	**3.285**
ILRT Dose Rate (person-rem/yr) from 3a and 3b		2.49E-03		8.48E-03		1.31E-02	
Delta Total Dose Rate[(1)]	From 3 yr	---		5.79E-03		1.03E-02	
	From 10 yr	---		---		4.48E-03	
3b Frequency (LERF)		6.89E-09		2.37E-08		3.70E-08	
Delta 3b LERF	From 3 yr	---		1.68E-08		3.01E-08	
	From 10 yr	---		---		1.33E-08	
CCFP %		86.25%		86.82%		87.28%	
Delta CCFP %	From 3 yr	---		0.57%		1.03%	
	From 10 yr	---		---		0.46%	

Note to Table 5.6-1:

(1) The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

TABLE 5.6-2

QCNPS ILRT EXTENSION RESULTS COMPARISON TO ACCEPTANCE CRITERIA

FIGURE OF MERIT - >	ΔLERF	ΔPERSON-REM/YR	ΔCCFP
QCNPS	3.0E-8/yr	1.0E-02/yr (0.31%)	1.0%
Acceptance Criteria	**<1.0E-7/yr (“very small”)**	**<1.0 person-rem/yr or <1.0%**	**<1.5%**

5.7 EXTERNAL EVENTS CONTRIBUTION

Since the risk acceptance guidelines in RG 1.174 are intended for comparison with a full-scope assessment of risk, including internal and external events, a bounding analysis of the potential impact from external events is presented here.

External hazards were evaluated in the QCNPS Individual Plant Examination of External Events (IPEEE) submittal in response to the NRC IPEEE Program (Generic Letter 88-20, Supplement 4) [20]. The IPEEE Program was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks.

The results of the QCNPS IPEEE study are documented in the QCNPS IPEEE Main Report [21] and related correspondence. The primary areas of external event evaluation at QCNPS were internal fire and seismic. The internal fire events were addressed by using the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology [22] and the guidance provided in the EPRI Fire PRA Implementation Guide [32]. The seismic evaluations were performed in accordance with the EPRI Seismic Margins Analysis (SMA) methodology [23].

The IPEEE seismic evaluation did not result in CDF or LERF results. Bounding seismic CDF values from the NRC have been made public as part of the development of a generic issue report. Referencing the Risk Assessment for NRC GI-199 [28], Table D-1 lists the postulated core damage frequencies using the updated 2008 USGS Seismic Hazard Curves. The weakest link model using the curve for QCNPS resulted in a CDF of 2.70E-05/yr. This value is utilized for the bounding external events assessment provided here.

In addition to internal fires and seismic events, the QCNPS IPEEE Submittal analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Flooding
- Transportation and Nearby Facility Accidents
- Other External Hazards

The QCNPS IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that QCNPS meets the applicable Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards. As such, these hazards were determined in the QCNPS IPEEE to be negligible contributors to overall plant risk.

Accordingly, these other external event hazards are not included explicitly in this section and are reasonably assumed not to impact the results or conclusions of the ILRT interval extension risk assessment.

5.7.1 QCNPS Fire Risk Discussion

A quantifiable Fire PRA model meeting an appropriate level of ASME/ANS Standard [19] is under development for QCNPS. The QCNPS Fire PRA updated in 1999 as part of the revised QCNPS Individual Plant Examination of External Events (IPEEE) Submittal is judged to be adequate to support the ILRT External Events quantitative risk assessment.

While the IPEEE fire analysis did yield a CDF, the intent of the analysis was to identify the most risk significant fire areas in the plant using a screening process and by calculating conservative core damage frequencies for fire scenarios. The screening attributes of the fire PRA are summarized below.

Attributes of Fire PRA

Fire PRAs are useful tools to identify design or procedural items that could be clear areas of focus for improving the safety of the plant. Fire PRAs use a structure and quantification technique similar to that used in the internal events PRA.

Historically, since less attention has been paid to fire PRAs, conservative modeling is common in a number of areas of the fire analysis to provide a "bounding" methodology for fires. This concept is contrary to the base internal events PRA which has had more analytical development and is closer to a realistic assessment (i.e., not conservative) of the plant.

There are a number of fire PRA topics involving technical inputs, data, and modeling that prevent the effective comparison of the calculated core damage frequency figure of merit between the internal events PRA and the fire PRA. These areas are identified as follows:

- **Initiating Events**: The frequency of fires and their severity are generally conservatively overestimated. A revised NRC fire events database indicates the trend toward both lower frequency and less severe fires. This trend reflects the improved housekeeping, reduction in transient fire hazards, and other improved fire protection steps at nuclear utilities. The database used in the QCNPS fire assessment used significantly older data that is conservative compared to more current data.
- **System Response**: Fire protection measures such as sprinklers, CO_2, and fire brigades may be given minimal (conservative) credit in their ability to limit the spread of a fire. Therefore, the severity of the fire and its impact on requirements is exacerbated.

  In addition, cable routings are typically characterized conservatively because of the lack of data regarding the routing of cables or the lack of the analytic modeling to represent the different routings. This leads to limited credit for balance of plant systems that are extremely important in CDF mitigation.

- **Sequences**: Sequences may subsume a number of fire scenarios to reduce the analytic burden. The subsuming of initiators and sequences is done to envelope those sequences included. This causes additional conservatism.
- **Fire Modeling**: Fire damage and fire propagation are conservatively characterized. Fire modeling presents bounding approaches regarding the fire immediate effects (e.g., all cables in a tray are always failed for a cable tray fire) and fire propagation.

The fire PRA is subject to more modeling uncertainty than the internal events PRA evaluations. While the fire PRA is generally self-consistent within its calculational framework, the fire PRA calculated quantitative risk metric does not compare well with internal events PRAs because of the number of conservatisms that have been included in the fire PRA process. Therefore, the use of the fire PRA figure of merit as a reflection of CDF may be inappropriate. Any use of fire PRA results and insights should properly reflect consideration of the fact that the “state of the technology” in fire PRAs is less evolved than the internal events PRA.

Relative modeling uncertainty is expected to narrow substantially in the future as more experience is gained in the development and implementation of methods and techniques for modeling fire accident progression and the underlying data.

The QCNPS risk due to internal fires was updated in 1999 as part of the revised QCNPS Individual Plant Examination of External Events (IPEEE) Submittal. The EPRI FIVE Methodology and Fire PSA Implementation Guide screening approaches and data were used to perform the study.

The QCNPS Unit 1 CDF contribution due to internal fires in the unscreened fire areas was calculated at 6.60E-5/yr. The breakdown of the QCNPS fire risk profile is as follows [29]:

TABLE 5.7-1

QCNPS UNIT 1 FIRE RISK PROFILE

RELEASE TYPE	CONTRIBUTION	CDF
Fire-induced loss of decay heat removal scenarios	80.4%	5.31E-05
Fire-induced loss of inventory control scenarios (RPV at low pressure)	4.3%	2.84E-06
Fire-induced loss of inventory control scenarios (RPV at high pressure)	3.9%	2.57E-06
Other fire-induced scenarios	11.4%	7.52E-06
TOTALS	**100%**	**6.60E-05**

Per the NEI Guidance, the impact on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

Delta LERF = (Frequency of EPRI Category 3b for 1-per-15 year ILRT interval) – (Frequency of EPRI Category 3b for 1-per-10 year ILRT interval)

As discussed in Section 4.3, the frequency per year for EPRI Category 3b is calculated as:

Frequency 3b = [3b conditional failure probability] x [CDF - (CDF with independent LERF + CDF that cannot cause LERF)]

The following external event accident scenario is treated separately in the 3b frequency calculation because decay heat removal has a lower probability in leading to a LERF release. The probability of a LERF Release from Class II events in the FPIE PRA model is 4.17%, as shown in Table 5.7-2. The LERF release is primarily due to a failure to declare a General Emergency early during a Loss of Decay Heat Removal scenario. For the Class II scenarios where General Emergency is declared early, the core damage event would lead to a release in the intermediate time frame rather than the early time frame such that the release would not lead to LERF. These Class II scenarios where General Emergency is declared early will be removed from both Fire CDF and FPIE CDF results. The remaining Fire CDF results are assumed to align more closely with FPIE CDF results and a multiplier approach will be used to determine the increase in LERF and dose due to ILRT test surveillance change from external events.

TABLE 5.7-2

QC FPIE LERF CONTRIBUTION FROM DECAY HEAT REMOVAL SEQUENCES

CLASS II CALCULATION(1)	
Class II FPIE H/E Release Frequency	4.30E-08 /yr.
Class II FPIE Total Release Frequency	1.03E-06 /yr
Percent Class II H/E Release	4.17%

Note to Table 5.7-2:

(1) Release frequencies are from Table 4.2-3.

The table above shows that the probability of a release from a Class II release scenario is 4.17%. This value is rounded up to 5%, consistent with the FPIE event PRA model basic event B--OPDHR-EAL2F-- 'General Emergency Declared Late during Loss of Decay Heat Removal. This basic event has a failure probability of 0.05, or 5%. With this background, it is assumed that 95% of Class II heat removal scenarios cannot lead to LERF as early emergency evacuation results in a non-LERF scenario. This 5% is assumed applicable to the Fire PRA model results as well.

As outlined in Table 5.7-1, fire-induced loss of decay heat removal scenarios is 5.31E-5/yr. Assuming that 95% of the fire-induced loss of decay heat removal scenarios have the GE declared 'early", the fire induced release frequency that cannot be LERF is:

5.31E-5/yr. * 0.95 = 5.04E-5/yr.

The Fire CDF applicable to 3b frequency calculation is therefore,

Total CDF – Decay Heat Removal CDF = 6.60E-05/yr. – 5.04E-5/yr
Fire CDF applicable to 3b frequency = 1.56E-05/yr.

As outlined in Table 4.2-3, the Full Power Internal Events (FPIE) CDF is 2.92E-06/yr, with a Class II CDF contribution of 1.03E-06/yr.

Assuming that 95% of the FPIE loss of decay heat removal scenarios have the GE declared ‘early”, the FPIE Class II release frequency that cannot be LERF is:

1.03E-6/yr. * 0.95 = 9.79E-7/yr.

To compare the Fire CDF with the FPIE CDF, the class II sequence contribution not leading to an ‘early’ release is removed.

Total FPIE CDF – non-LERF Class II FPIE CDF = 2.92E-06/yr. – 9.79E-07/yr
FPIE CDF applicable to 3b frequency = 1.94E-06/yr.

Since ILRT delta risk is primarily a function of CDF, it could be assumed that the total impact from the ILRT fire risk contribution is bounded by assuming a fire multiplier factor of 8.0 (1.56E-05/yr ÷ 1.94E-06/yr) compared to the internal events evaluation alone. However, the fire multiplier will be applied towards the full FPIE CDF (2.92E-06/yr) which includes Class II non-LERF sequences. To better represent the impact of fire risk for the ILRT assessment where non-LERF frequencies can be excluded, the fire multiplier of 8.0 can be discounted by the ratio of FPIE CDF with Class II sequences removed to the full FPIE CDF (i.e., 1.94E-06/yr ÷ 2.92E-06/yr = 0.665) to give a fire multiplier of 5.3.

LERF was not quantified in the IPEEE; therefore a LERF estimate must also be developed. The internal events LERF value for the QC FPIE model is 1.97E-07/yr. Consistent with the approach for CDF, the QC Fire LERF is assumed to be a factor of 8.0 higher than the FPIE model LERF.

Fire LERF = FPIE LERF (1.97E-07 /yr.) x 8.0 (Fire CDF w/o Class II non-LERF contributions) ÷ FPIE CDF (w/o Class II non-LERF contributions))
Fire LERF = 1.58E-06/yr.

Note that the FPIE LERF value of 1.97E-07/yr includes the Class II LERF contribution (i.e., 4.17%) and therefore, the Fire LERF estimate developed by this ratio approach also includes the Class II contribution from a release with late declaration of a GE. The FPIE LERF value does not include Class II CDF with early declaration of a GE (i.e., 95.8%). Therefore, the approach used above is appropriate for estimating a Fire LERF value (i.e., Class II contributions do not need to further be subtracted).

5.7.2 QCNPS Seismic Risk Discussion

A quantifiable seismic PRA model for QCNPS has not yet been approved for general use in risk applications. However, recent information is available from the NRC. A Risk Assessment for NRC GI-199 "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States (CEUS) on Existing Plants," [28], Table D-1 lists the postulated core damage frequencies using the updated 2008 USGS Seismic Hazard Curves. For QCNPS, the Seismic Hazard CDF using the "Weakest Link" is 2.7E-05/yr. Given that seismic CDF contributions (e.g., by accident class) are not available, seismic LERF will be estimated by assuming the FPIE CDF contributions to LERF also apply to the seismic LERF.

For CDF, the seismic CDF is a factor of 9.25 greater than FPIE CDF (i.e., 2.7E-05 / 2.92E-06). Given this, it is reasonable to assume that the total CDF impact from seismic risk can be approximated by assuming a factor of 9.25 additional contribution to CDF compared to the internal events evaluation alone. Using a base FPIE LERF value of 1.97E-07/yr and multiplying by 9.25 for seismic, gives a total LERF estimate for the seismic PRA model of 1.82E-06/yr.

The assumptions regarding the CDF and LERF values provided above are used to provide insight into the impact of the total external hazard risk on the conclusions of this ILRT risk assessment.

5.7.3 Other External Events Discussion

In addition to internal fires and seismic events, the QC IPEEE Submittal analyzed a variety of other external hazards:

- High Winds/Tornadoes
- External Floods

- Transportation and Nearby Facility Accidents

The QCNPS IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, and nearby facility accidents was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based upon this review, it was concluded that QCNPS meets the applicable NRC Standard Review Plan requirements and therefore has an acceptably low risk with respect to these hazards.

Based on the other external events being low risk contributors (compared to fire and seismic events) the increase in the QCNPS other external events risk due to the ILRT extension is reasonably assumed to not impact the results or conclusions of the risk assessment.

5.7.4 External Events Impact Summary

In summary, the seismic and fire CDF values described above result in an external events bounding risk estimate of 9.3E-05/yr. Seismic and Fire LERF values derived from CDF values in sections 5.7.2 and 5.7.3 and as shown in Table 5.7-3 below sum to LERF value of 3.4E-06/yr, which is 17.3 times higher than the internal events LERF.

Table 5.7-3 summarizes the estimated bounding external events CDF contribution for QCNPS.

TABLE 5.7-3

QCNPS EXTERNAL EVENTS CONTRIBUTOR SUMMARY

EXTERNAL EVENT INITIATOR GROUP	CDF (1/YR)	LERF (1/YR)
Fire	6.6E-05	1.6E-06
Seismic	2.7E-05	1.86E-06
High Winds	Screened	Screened
Other Hazards	Screened	Screened
Total For External Events (for initiators with CDF/LERF available)	**9.3E-05**	**3.4E-06**
Internal Events	2.9E-06	2.0E-07

As noted earlier, the 3b contribution is approximately proportional to CDF. An increase in CDF would likely lead to higher 3b frequency and assumed LERF. The Fire CDF contributors were adjusted to remove Class II scenarios where an early declaration of a General Emergency was declared. The sequence contribution for the Seismic CDF is unknown and no adjustments were made. To determine a suitable multiplier of external CDF to internal event CDF, a multiplier is developed for each external event group (i.e., fire and seismic) and then added together to address both contributors, as shown in Table 5.7-4. For fire contribution, the adjusted CDF (i.e., Class II scenarios removed) ratio of fire and FPIE is multiplied by the portion of FPIE CDF that the fire contribution can act upon (i.e., ratio of adjusted FPIE CDF and unadjusted FPIE CDF). For seismic, the ratio of unadjusted CDF (i.e., seismic and FPIE) is used.

TABLE 5.7-4

QC EXTERNAL EVENTS TO INTERNAL EVENTS CDF COMPARISON

EVENT INITIATOR GROUP	CDF (1/YR)	ADJUSTED CDF (1/YR)	EVENT INITIATOR GROUP	ADJUSTED CDF (1/YR)	INITIAL MULTIPLIER	APPLICABLE MULTIPLIER PORTION(2)
Fire	6.6E-5	1.6E-5(1)	FPIE (reduced Class II frequency)	1.9E-6(1)	8.0	5.3
Seismic	2.7E-5	N/A	FPIE (full Class II frequency)	2.9E-6	9.3	9.3
External Event CDF to FPIE CDF Multipler					**14.6**	

Note to Table 5.7-4:

(1) 95% of Class II CDF contribution removed from Fire CDF as discussed previously, and also the Full Power Internal Events CDF to develop a multiplier for Fire CDF/FPIE CDF.

(2) The initial fire multiplier is reduced by a factor of 0.665 (i.e., 1.94E-06/yr / 2.92E-06/yr) because the intial fire multiplier is only applicable to a portion of the unadjusted FPIE CDF (2.92E-06/yr). The initial seismic multiplier is based on the unadjusted FPIE CDF and therefore no further reduction factor is applied.

5.7.5 External Events Impact on ILRT Extension Assessment

The EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT intervals are shown in Table 5.6-1 as 6.9E-09/yr, 2.4E-08/yr, and 3.7E-08/yr, respectively. Using an external events LERF multiplier of 14.6 (multiplier from Table 5.7-4) for QCNPS, the change in the LERF risk measure due to extending the ILRT from 3-per-10 years to 1-per-15 years, including both internal and external hazards risk, is estimated as shown in Table 5.7-5.

TABLE 5.7-5

QCNPS 3B (LERF/YR) AS A FUNCTION OF ILRT FREQUENCY FOR INTERNAL AND EXTERNAL EVENTS

(INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

	3B FREQUENCY (3-PER-10 YR ILRT)	3B FREQUENCY (1-PER-10 YEAR ILRT)	3B FREQUENCY (1-PER-15 YEAR ILRT)	LERF INCREASE(1)
Internal Events Contribution	6.9E-09	2.4E-08	3.7E-08	3.0E-08
External Events Contribution (Internal Events x 14.6)	1.0E-07	3.5E-07	5.4E-07	4.4E-07
Combined (Internal + External)	**1.1E-07**	**3.7E-07**	**5.8E-07**	**4.7E-07**

Note to Table 5.7-5:

(1) Associated with the change from the baseline 3-per-10 year frequency to the proposed 1-per-15 year frequency.

The other metrics for the ILRT extension risk assessment can be similarly derived using the multiplier approach. The results between the 3-in-10 year interval and the 15 year interval compared to the acceptance criteria are shown in Table 5.7-6. As can be seen, the impact from including the external events contributors would not change the conclusion of the risk assessment. That is, the acceptance criteria are all met such that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years has been demonstrated to be small. Note that a bounding analysis for the total LERF contribution follows Table 5.7-6 to demonstrate that the total LERF value for QCNPS is less than 1.0E-05/yr consistent with the requirements for a “Small Change” in risk of the RG 1.174 acceptance guidelines.

TABLE 5.7-6

COMPARISON TO ACCEPTANCE CRITERIA INCLUDING EXTERNAL EVENTS CONTRIBUTION FOR QCNPS

CONTRIBUTOR	∆LERF	∆PERSON-REM/YR	∆CCFP[1]
Internal Events	3.0E-08	1.0E-02 (0.31%)	1.03%
External Events	4.4E-07	1.5E-01[2] (0.31%)	1.03%
Total	**4.7E-07**	**1.6E-01 (0.31%)**	**1.03%**
Acceptance Criteria	**<1.0E-6/yr (“small”)**	**<1.0 person-rem/yr <u>or</u> <1.0%**	**<1.5%**

<u>Notes to Table 5.7-6</u>:

(1) The probability of leakage due to the ILRT extension is assumed to be the same for both Internal and External events. Therefore, the percentage change for CCFP remains constant.

(2) Calculated as the FPIE value times the external events multiplier of 14.6 developed in Table 5.7-4.

The 4.7E-07/yr increase in LERF due to the combined internal and external events from extending the ILRT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1.0E-7 to 1.0E-6 per reactor year (“small” change in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the “small” change range, the risk assessment must also reasonably show that the total LERF is less than 1.0E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above are used for the total LERF estimate.

From Table 4.2-2, the LERF (High Early) due to postulated internal event accidents is 2.0E-07/yr for QCNPS. As discussed in Sections 5.7.1 and shown in Table 5.7-3, the LERF estimate for the Fire PRA model is 1.6E-06 /yr. As discussed in Sections 5.7.2 and shown in Table 5.7-3, the total LERF estimate for the Seismic PRA model is 1.8E-06 /yr. The total LERF values for QCNPS are shown in Table 5.7-7.

TABLE 5.7-7

IMPACT OF 15-YR ILRT EXTENSION ON LERF FOR QCNPS

LERF CONTRIBUTOR	(1/YR)
Internal Events LERF	2.0E-07
Fire LERF	1.6E-06
Seismic LERF	1.8E-06
Internal Events LERF due to ILRT (at 15 years) (1)	3.7E-08
External Events LERF due to ILRT (at 15 years) (1)	5.4E-07 [Internal Events LERF due to ILRT * 14.6]
Total	**4.2E-06/yr**

Note to Table 5.7-7:

(1) Including age adjusted steel liner corrosion likelihood as reported in Table 5.7-5.

As can be seen, the estimated upper bound LERF for QCNPS is estimated as 4.2E-06/yr. This value is less than the RG 1.174 requirement to demonstrate that the total LERF due to internal and external events is less than 1.0E-05/yr.

5.8 CONTAINMENT OVERPRESSURE IMPACTS ON CDF

As indicated in the EPRI ILRT report [3], in general, CDF is not significantly impacted by an extension of the ILRT interval. However, plants that rely on containment overpressure for net positive suction head (NPSH) for emergency core coolant system (ECCS) injection for certain accident sequences may experience an increase in CDF.

For QCNPS, there is some dependency on NPSH. The QCNPS PRA model does include scenarios where CDF could be impacted due to an increase in the likelihood for a loss of containment overpressure resulting from a pre-existing leak from containment and loss of heat removal systems. Per the EPRI guidance, as a first order estimate of the impact, it can be assumed that the EPRI Class 3b contribution would lead to loss of containment overpressure. For the QC PRA model, applying that guidance would mean that the current containment isolation failure logic can be increased by the Class 3b frequency at 15 years (i.e., 0.0023 * 5.0 = 0.0115) to estimate a bounding increase in CDF. With this increase applied to the containment isolation failure probability in the QC PRA model, the CDF increases from 2.921E-06 /yr to 2.993E-06 /yr representing an increase of just 7.2E-08 /yr. The bounding analysis is in the very small range (CDF <1E-06 /yr) per RG-1.174, and as such the focus on the LERF figure of merit for this application is appropriate for QCNPS.

6.0 SENSITIVITIES

6.1 SENSITIVITY TO CORROSION IMPACT ASSUMPTIONS

The results in Tables 5.2-2, 5.3-1, and 5.3-2 show that including corrosion effects calculated using the assumptions described in Section 4.4 does not significantly affect the results of the ILRT extension risk assessment. In any event, sensitivity cases were developed to gain an understanding of the sensitivity of the results to the key parameters in the corrosion risk analysis. The time for the flaw likelihood to double was adjusted from every five years to every two and every ten years. The failure probabilities for the wall and basemat were increased and decreased by an order of magnitude. The total detection failure likelihood was adjusted from 10% to 15% and 5%. The results are presented in Table 6.1-1. In every case, the impact from including the corrosion effects is minimal. Even the upper bound estimates with conservative assumptions for all of the key parameters yield increases in LERF due to corrosion of only 1.4E-07/yr. The results indicate that even with conservative assumptions, the conclusions from the base analysis would not significantly change.

TABLE 6.1-1

STEEL LINER CORROSION SENSITIVITY CASES FOR QCNPS

AGE (STEP 3 IN THE CORROSION ANALYSIS)	CONTAINMENT BREACH (STEP 4 IN THE CORROSION ANALYSIS)	VISUAL INSPECTION & NON-VISUAL FLAWS (STEP 5 IN THE CORROSION ANALYSIS)	INCREASE IN CLASS 3B FREQUENCY (LERF) FOR ILRT EXTENSION FROM 3 IN 10 TO 1 IN 15 YEARS (PER YEAR)	
			TOTAL INCREASE	INCREASE DUE TO CORROSION
Base Case Doubles every 5 yrs	Base Case (10% Wall, 1% Basemat)	Base Case (10% Wall, 100% Basemat)	3.01E-08	3.73E-09
Doubles every 2 yrs	Base	Base	3.49E-08	8.53E-09
Doubles every 10 yrs	Base	Base	2.95E-08	3.15E-09
Base	Base	15% Wall	3.13E-08	4.97E-09
Base	Base	5% Wall	2.88E-08	2.49E-09
Base	100% Wall, 10% Basemat	Base	6.36E-08	3.73E-08
Base	1.0% Wall, 0.1% Basemat	Base	2.67E-08	3.73E-10
LOWER BOUND				
Doubles every 10 yrs	1.0% Wall, 0.1% Basemat	5% Wall, 100% Basemat	2.65E-08	2.10E-10
UPPER BOUND				
Doubles every 2 yrs	100% Wall, 10% Basemat	15% Wall, 100% Basemat	1.40E-07	1.14E-07

6.2 EPRI EXPERT ELICITATION SENSITIVITY

An expert elicitation was performed by EPRI [3] to reduce excess conservatisms in the data associated with the probability of undetected leaks within containment. Since the risk impact assessment of the extensions to the ILRT interval is sensitive to both the probability of the leakage as well as the magnitude, it was decided to perform the expert elicitation in a manner to solicit the probability of leakage as a function of leakage magnitude. In addition, the elicitation was performed for a range of failure modes which allowed experts to account for the range of failure mechanisms, the potential for undiscovered mechanisms, inaccessible areas of the containment as well as the potential for detection by alternate means. 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 basic difference in the application of the ILRT interval methodology using the expert elicitation is a change in the probability of pre-existing leakage within containment. The base case methodology uses the Jeffrey's non-informative prior for the large leak size and the expert elicitation sensitivity study uses the results from 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 base case methodology (i.e., 10 La for small and 100 La for large) are used here. Table 6.2-1 illustrates the magnitudes and probabilities of a pre-existing leak in containment associated with the base case and the expert elicitation statistical treatments. These values are used in the ILRT interval extension for the base methodology and in this sensitivity case. Details of the expert elicitation process, including the input to expert elicitation as well as the results of the expert elicitation, are available in the various appendices of EPRI 1018243 [3].

TABLE 6.2-1

EPRI EXPERT ELICITATION RESULTS

LEAKAGE SIZE (LA)	BASE CASE MEAN PROBABILITY OF OCCURRENCE	EXPERT ELICITATION MEAN PROBABILITY OF OCCURRENCE [3]	PERCENT REDUCTION
10	9.2E-03	3.9E-03	58%
100	2.3E-03	2.5E-04	89%

The summary of results using the expert elicitation values for probability of containment leakage is provided in Table 6.2-2. As mentioned previously, probability values are those associated with the magnitude of the leakage used in the base case evaluation (10 La for small and 100 La for large). The expert elicitation process produces a relationship between probability and leakage magnitude in which it is possible to assess higher leakage magnitudes that are more reflective of large early releases; however, these evaluations are not performed in this particular study.

The net effect is that the reduction in the multipliers shown above also leads to a dramatic reduction on the calculated increases in the LERF values. As shown in Table 6.2-2, the increase in the overall value for LERF due to Class 3b sequences that is due to increasing the ILRT test interval from 3 to 15 years is just 6.6E-09/yr. Similarly, the increase due to increasing the interval from 10 to 15 years is just 3.5E-09/yr. As such, if the expert elicitation probabilities of occurrence are used instead of the non-informative prior estimates, the change in LERF is well within the range of a “very small” change in risk when compared to the current 1-in-10, or baseline 3-in-10 year requirement. Additionally, as shown in Table 6.2-2, the increase in dose rate and CCFP are similarly reduced to much smaller values. The results of this sensitivity study are judged to be more indicative of the actual risk associated with the ILRT extension than the results from the assessment as dictated by the values from the EPRI methodology [3], and yet are still conservative given the assumption that all of the Class 3b contribution is considered to be LERF.

TABLE 6.2-2

QCNPS ILRT CASES:
3 IN 10 (BASE CASE), 1 IN 10, AND 1 IN 15 YR INTERVALS
(BASED ON EPRI EXPERT ELICITATION LEAKAGE PROBABILITIES)

EPRI CLASS	DOSE PER-REM	BASE CASE 3 IN 10 YEARS		EXTEND TO 1 IN 10 YEARS		EXTEND TO 1 IN 15 YEARS	
		CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR	CDF (1/YR)	PERSON-REM/YR
1	2.62E+03	3.96E-07	1.04E-03	3.67E-07	9.61E-04	3.45E-07	9.03E-04
2	2.16E+06	1.07E-08	2.31E-02	1.07E-08	2.31E-02	1.07E-08	2.31E-02
3a	2.62E+04	1.11E-08	2.91E-04	3.70E-08	9.67E-04	5.55E-08	1.45E-03
3b	2.62E+05	1.01E-09	2.65E-04	4.10E-09	1.07E-03	7.57E-09	1.98E-03
7	1.25E+06	2.45E-06	3.06E+00	2.45E-06	3.06E+00	2.45E-06	3.06E+00
8	4.11E+06	4.70E-08	1.93E-01	4.70E-08	1.93E-01	4.70E-08	1.93E-01
Total		**2.92E-06**	**3.273**	**2.92E-06**	**3.274**	**2.92E-06**	**3.276**
ILRT Dose Rate from 3a and 3b		5.55E-04		2.04E-03		3.43E-03	
Delta Total Dose Rate(1)	From 3 yr	---		1.41E-03		2.74E-03	
	From 10 yr	---		---		1.34E-03	
3b Frequency (LERF)		1.01E-09		4.10E-09		7.57E-09	
Delta 3b LERF	From 3 yr	---		3.09E-09		6.56E-09	
	From 10 yr	---		---		3.47E-09	
CCFP %		86.05%		86.15%		86.27%	
Delta CCFP %	From 3 yr	---		0.11%		0.22%	
	From 10 yr	---		---		0.12%	

Note to Table 6.2-2:

(1) The overall difference in total dose rate is less than the difference of only the 3a and 3b categories between two testing intervals. This is due to the fact that the Class 1 person-rem/yr decreases when extending the ILRT frequency.

7.0 CONCLUSIONS

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

- Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines "very small" changes in risk as resulting in increases of CDF below 1.0E-06/yr and increases in LERF below 1.0E-07/yr. "Small" changes in risk are defined as increases in CDF below 1.0E-05/yr and increases in LERF below 1.0E-06/yr. Since the ILRT extension was demonstrated to have negligible impact on CDF for QCNPS, the relevant criterion is LERF. The increase in internal events LERF resulting from a change in the Type A ILRT test interval for the base case with corrosion included is 3.0E-08/yr (see Table 5.6-1). In using the EPRI Expert Elicitation methodology, the change is estimated as 6.6E-09/yr (see Table 6.2-2). Both of these values fall within the "very small" change region of the acceptance guidelines in Reg. Guide 1.174.
- The change in dose risk for changing the Type A test frequency from three-per-ten years to once-per-fifteen-years, measured as an increase to the total integrated dose risk for all internal events accident sequences for QCNPS, is 1.0E-02 person-rem/yr (0.31%) using the EPRI guidance with the base case corrosion included (Table 5.6-1). The change in dose risk drops to 2.7E-03 person-rem/yr (0.08%) when using the EPRI Expert Elicitation methodology (Table 6.2-2). The values calculated per the EPRI guidance are all lower than the acceptance criteria of ≤1.0 person-rem/yr or <1.0% person-rem/yr defined in Section 1.3.
- The increase in the conditional containment failure frequency from the three in ten year interval to one in fifteen years including corrosion effects using the EPRI guidance (see Section 5.5) is 1.0%. This value drops to 0.22% using the EPRI Expert Elicitation methodology (see Table 6.2-2). Both of these values are below the acceptance criteria of less than 1.5% defined in Section 1.3.
- To determine the potential impact from external events, a bounding assessment from the risk associated with external events was performed utilizing available information. As shown in Table 5.7-6, the total increase in LERF due to internal events and the bounding external events assessment is 4.7E-07/yr. This value is in Region II of the Reg. Guide 1.174 acceptance guidelines ("small" change in risk). The changes in dose risk and conditional containment failure frequency also remained below the acceptance criteria.

- As shown in Table 5.7-7, the same bounding analysis indicates that the total LERF from both internal and external risks is 4.2E-06/yr which is less than the Reg. Guide 1.174 limit of 1.0E-05/yr given that the ΔLERF is in Region II ("small" change in risk).
- Including age-adjusted steel liner corrosion effects in the ILRT assessment was demonstrated to be a small contributor to the impact of extending the ILRT interval for QCNPS.

Therefore, 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 QCNPS risk profiles.

Previous Assessments

The NRC in NUREG-1493 [6] has previously concluded the following:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one 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 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, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated. Beyond testing the performance of containment penetrations, ILRTs also test the integrity of the containment structure.

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

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APPENDIX A
PRA TECHNICAL ADEQUACY

A.1 OVERVIEW

A Probabilistic Risk Assessment (PRA) analysis is presented in this report to support an extension of the QCNPS Unit 1 and Unit 2 containment Type A Integrated Leak Rate Test (ILRT) interval to fifteen years.

The analysis follows the guidance provided in Regulatory Guide 1.200, Revision 2 [A.1], "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The guidance in RG-1.200 indicates that the following steps should be followed to perform this study:

1. Identify the parts of the PRA used to support the application
   - SSCs, operational characteristics affected by the application and how these are implemented in the PRA model.
   - A definition of the acceptance criteria used for the application.
2. Identify the scope of risk contributors addressed by the PRA model
   - If not full scope (i.e. internal and external), identify appropriate compensatory measures or provide bounding arguments to address the risk contributors not addressed by the model.
3. Summarize the risk assessment methodology used to assess the risk of the application
   - Include how the PRA model was modified to appropriately model the risk impact of the change request.
4. Demonstrate the Technical Adequacy of the PRA
   - Identify plant changes (design or operational practices) that have been incorporated at the site, but are not yet in the PRA model and justify why the change does not impact the PRA results used to support the application.
   - Document peer review findings and observations that are applicable to the parts of the PRA required for the application, and for those that have not yet been addressed justify why the significant contributors would not be impacted.
   - Document that the parts of the PRA used in the decision are consistent with applicable standards endorsed by the Regulatory Guide. Provide justification to show that where specific requirements in the standard are not met, it will not unduly impact the results.
   - Identify key assumptions and approximations relevant to the results used in the decision-making process.

Items 1 through 3 are covered in the main body of this report. The purpose of this appendix is to address the requirements identified in item 4 above. Each of these item 4 aspects (i.e., plant changes not yet incorporated into the PRA model, relevant peer review findings, consistency with applicable PRA standards, and the identification of key assumptions) are discussed in the following sections.

The risk assessment performed for the ILRT extension request is based on the current Level 1 and Level 2 PRA model. Note that for this application, the accepted methodology involves a bounding approach to estimate the change in the LERF from extending the ILRT interval. Rather than exercising the PRA model itself, it involves the establishment of separate evaluations that are linearly related to the plant CDF contribution. Consequently, a reasonable representation of the plant CDF that does not result in a LERF does not require that Capability Category II be met in every aspect of the modeling if the Category I treatment is conservative or otherwise does not significantly impact the results.

A discussion of the Exelon model update process, the peer reviews performed on the QCNPS models, the results of those peer reviews and the potential impact of peer review findings on the ILRT extension risk assessment are provided in Section A.2. Section A.3 provides an assessment of key assumptions and approximations used in this assessment. Finally, Section A.4 briefly summarizes the results of the PRA technical adequacy assessment with respect to this application.

A.2 PRA MODEL EVOLUTION AND PEER REVIEW SUMMARY

A.2.1 Introduction

The 2014A versions of the QCNPS PRA models are the most recent evaluations of the Unit 1 and Unit 2 risk profile at QCNPS for internal event challenges. The QCNPS 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 QCNPS 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. The following information describes this approach as it applies to the QCNPS PRA.

A.2.2 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 and subordinate implementation procedures. The PRA model update procedure delineates the responsibilities and guidelines for updating the full power internal events PRA models at all operating Exelon nuclear generation sites. The overall Exelon Risk Management program defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, industry operating experience, etc.), and for 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:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on CDF is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities are updated approximately every four years.

In addition to these activities, Exelon risk management procedures provide the guidance for particular risk management maintenance activities. This guidance includes:

- 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.
- 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)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant. The 2014A models were completed in January of 2015.

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 the relevant items (i.e., plant changes not yet incorporated into the PRA model, relevant peer review findings, and consistency with applicable PRA Standards) are discussed in turn in this section.

A.2.3 Plant Changes Not Yet Incorporated into the PRA Model

A PRA updating requirements evaluation (URE- Exelon PRA model update tracking database) is created for all issues that are identified that could impact the PRA model. The URE database includes the identification of those plant changes that could impact the PRA model.

A review of the open UREs indicates that there are no plant changes that have not yet been incorporated into the PRA model that would affect this application. FLEX modifications are in progress and will be incorporated in the QC PRA in the future. The FLEX strategy will reduce CDF and is expected to lead to a reduction in the risk associated with the proposed ILRT extension. At this time, there is insufficient information to quantify the impact to this application, but the omission of FLEX credit in the model should in general result in added conservatism to the ILRT results.

A.2.4 Consistency with Applicable PRA Standards

Several assessments of technical capability have been made for the QCNPS internal events PRA models. These assessments are as follows and are further discussed in the paragraphs below.

- An independent PRA peer review [A.7] was conducted under the auspices of the BWR Owners Group in 2000, following the Industry PRA Peer Review process [A.2]. This peer review included an assessment of the PRA model maintenance and update process.
- In 2004, a gap analysis was performed to assess gaps between the peer review scope/detail of the Industry PRA Peer Review results relative to the available version of the ASME PRA Standard [A.3] and the draft version of Regulatory Guide 1.200, DG-1122 [A.4].
- During 2005 and 2006 the QCNPS PRA model results were evaluated in the BWR Owners Group PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process [A.5].
- In January 2010, a self-assessment analysis was performed against the available version of the ASME/ANS PRA Standard [A.6] in preparation for the QCNPS 2010 PRA periodic update.
- In May 2010, an independent Focused PRA Peer Review [A.9] of the QCNPS Internal Flooding (IF) PRA model was performed using the NEI 05-04 process [A.16], the ASME/ANS PRA Standard [A.6], and RG 1.200, Rev. 2 [A.1]. The results of this IF assessment are used as the basis for the capability assessment provided in Table A-1.
- The QC 2010 self-assessment [A.10] was updated to incorporate the results of the final Focused PRA Peer Review report of the Internal Flooding PRA model.

- Following the most recent 2014 PRA update, another self-assessment [A.11] was performed to reflect the status after the 2014A model. This self-assessment was performed against the ASME/ANS PRA Standard [A.6], and RG 1.200, Rev. 2 [A.1].
- In February 2017, an independent PRA Peer Review [A.15] of the QC Internal Events PRA model was performed using the NEI 05-04 Rev. 2 [A.16] process, the ASME PRA Standard [A.6], and RG 1.200, Rev. 2 [A.1]. The Peer review included all SRs except those related to internal flooding (which was previously peer reviewed in 2010). In addition, four SRs were assessed as not applicable to the Quad Cities PRA. The results of this assessment are used as the basis for the capability assessment provided in Table A-2.

With the 2010 IF and 2017 Peer Reviews, all elements of the QCNPS PRA have undergone a thorough PRA Peer Review against the PRA Standard [A.6] and RG 1.200 Rev. 2 [A.1]. The results of the 2017 PRA Peer Review are as follows:

- SR Capability
  - 92% of the 259 applicable SRs are graded at Capability Category II or greater
  - 3% of the SRs are graded at Capability Category I
  - 5% of the SRs are graded as “Not Met”
- Facts and Observations
  - There were 31 Findings. Table A-2 provides an assessment of each finding with respect to the potential impact on the ILRT application.

A.2.5 <u>Applicability of Peer Review Findings and Observations</u>

Per the NRC SE [A.17] the appropriate PRA quality to support an ILRT risk assessment is that the PRA Standard Supporting Requirements should meet Capability Category I or greater. There are 316 Technical Supporting Requirements plus 10 Maintenance and Update Supporting Requirements in the FPIE portion of the ASME/ANS PRA Standard [A.6].

The 2010 Focused PRA Peer Review resulted in 3 findings, 10 suggestions and one best practice. Three supporting requirements were not met:

- IFSO-A3, IFSN-A7 and IFQU-A3

Table A-1 describes the findings associated with these IF SRs. The findings have been resolved and the findings have no impact on this application, as documented in Table A-1.

Per the 2017 QC PRA Peer Review [A.15], there are thirteen (13) Supporting Requirements that are not met:

- IE-C2, IE-C11, IE-C12, IE-D2, SY-A4, HR-G6, HR-G7, DA-C3, DA-C4, QU-B3, QU-C1, QU-E2, QU-E4

The 2017 QC PRA Peer Review identified seven (7) Supporting Requirements that are met at Capability Category I only:

- IE-B3, HR-D2, DA-D1, DA-D4, LE-C10, LE-C11, and LE-C12

As noted previously, Capability Category I is acceptable for an ILRT risk assessment.

The 2017 Peer Review findings are listed in Table A-2. The SRs associated with these findings are cross-referenced to the applicable findings within Table A-2. The 2017 Peer Review did not include a review of internal flooding SRs as this was performed in the 2010 Internal Flood focused peer review. The 2017 findings have not yet been resolved, but the potential impact upon the ILRT risk application results are assessed, as documented in Table A-2. No single finding was found to have a significant change to CDF if a model change was performed to address the finding. A number of finding resolutions will cause a small reduction in CDF. None of these findings are found to impact the conclusion of the ILRT risk application results. The cumulative impact of addressing all findings is judged to be minor and likely to reduce CDF.

TABLE A-1

2010 FPIE PEER IF FOCUSED REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IFQU-A3 IFSO-A3	Finding 1-2: IFQU-A3 and IFSO-A3 Not Met The quantitative screening criteria described in Section 2.3.6 and Section B.3 is less conservative than the screening criteria in the ASME/ANS PRA Standard. IFQU-A3 states that a flood area can be screened if the product of the sum of frequencies of flood scenarios for a flood area calculated based upon the bounding CCDP is less that 1E-9. Section 2.3.6 does not use the bounding CCDP but rather sums the CDFs for the floods. This would result in screening more areas out than would be screened using the criteria of this SR. The screening criterion of this SR is not used.	The QC IF has been updated to meet SR IFQU-A3 quantitative criteria including using the sum of the frequencies of the flood scenarios for the area, and (most bounding) CCDP less than 1E-9/yr. The Quad Cities Internal Flood PRA screening process also follows the qualitative process cited in the ASME/ANS PRA Standard in IFSO A1 & IFSO A2. IFSO A1 requires identification of four potential water source types. IFSO-A1 is now met. Examples include: "…circulation water system, service water systems, etc." – Circulating Water pipe leaks are significant contributors in the Turbine Building. Service Water piping leaks are considered in multiple areas. "tanks and pools located in the flood area" - Reactor Building equipment drain tank (RBEDT) located in the Reactor Building basement "..plant external sources of flooding (e.g., reservoirs or rivers)" – Examples: CST draining into the Reactor Building basement and RHRSW from the Ultimate Heat Sink draining into Reactor Building areas. "…in-leakage from other flood areas (e.g., backflow through drains, doorways, etc.)." – Ex: Backflow from SW to the RHR SW pump room sump area has been considered (3 check valves in series protect from backflow). Flow may be diverted to the torus room through an open hatch and Unit 1 flooding may be diverted through an open doorway to Unit 2. IFSO-A2 requires the inclusion of "sources with multi-unit or cross-unit impacts. A number of cross-unit and multi-unit scenarios are identified in the QC PRA model. Finding 1-2 has been addressed and there is no impact to this application.

TABLE A-1

2010 FPIE PEER IF FOCUSED REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IFSO-A1 IFSN-A1 IFSN-A8	Finding 1-6: IFSO-A1 and IFSN-A1 Met at CC I-III, IFSN-A8 Met at CC I-II Potential sources of flooding have been identified for each flood area, and include consideration of equipment such as pumps, tanks, piping, etc. Flooding from external sources and in-leakage from other flood areas was also considered. This is documented in Sections 2.3.3, Table B.3-3 of the IF Notebook. Check valve backflow failure rate is 1e-3/demand. QC does not currently rely on check valve modeling as RHRSW has 3 valves in series, and the corner rooms have a check valve and normally closed ball valve in series. However, the drain ball valves are routinely operated by operators to remove water from the corner room drains and could potentially be left open. Leak testing occurs once each 4 years. (see procedures QCOS 0020-04 and QCOS 0010-11). The potential for human error in leaving the ball valves open is not addressed in the model or documentation.	The potential for human error in leaving the ball valves open has been evaluated and found to be of negligible risk. Operators check the corner room drains every shift. Flood impacts to multiple corner rooms is required to significantly impact risk. Justification for not including these valves in the model has been added to the QC Internal Flooding Notebook (QC-PSA-012) Appendix D. In addition, references to backflow check valves in drain lines were deleted because these do not exist in QC corner room drain lines. Issue has been resolved. No impact to this application.
IFSN-A7	Finding 1-7: IFSN-A7 is Not Met The documentation of EQ equating to operability during a spray event does not meet the standard for expert judgment. Additionally, there is no data or analysis to justify this position. This was the only case identified in which this SR was used.	The description in 2.2.13.3 of the QC Internal Flooding Notebook (QC-PSA-012) is clarified to indicate that no credit is included for survivability of instruments that are sprayed. Issue has been resolved. No impact to this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IE-B3 IE-D1 IE-D3	Finding 1-2: IE-D1 (MET CC I-III), IE-B3 MET CC 1, IE-D2 (NOT MET) Section 2.5 of the IE notebook discussed the method used for IE categories. However, the CC-II requirements of SR IE-B3 are not explicitly reflected in the grouping steps, especially for the demonstration that such grouping does not impact significant accident sequences. Per staff, demonstration that the groupings does not impact significant accident sequences has not been completed for Quad Cities. This will need to be done in the next PRA update.	SR IE-D2 is not met. IE-D2 is a documentation only SR. SR IE-B3 requirements are currently met at CC I thereby supporting the ILRT risk assessment. This SR will be explicitely addressed in a future update. As noted in Section 2.5 of the Initiating Events Notebook "The initiating events are categorized into separate groups based on their different impacts on plant performance, safety functions, and possibilities of recovery. The categories or groups of initiators that are grouped together are events that are similar in their impact on plant response, including mitigating systems and timing, or are events that can be subsumed into a group by the worst case impacts within the group. Particular attention is paid to ensuring that initiators with significantly different impacts on LERF are grouped separately." The top 10 sequences contribute ~95% of CDF. These sequences include General Transients (Class IA), DLOOP (Class IIA), LOOP (Class IA), DLOOP (Class IBL), Small Break Water LOCA (Class IA), General Transient (Class IIA) , DLOOP (Class IBE) and Small Break Water LOCA (Class IIIC). The initiating event groupings entering the top 10 sequences were reviewed and found to be appropriately grouped. The groupings allow appropriate treatment using assigned event trees and initiator impacts with fault tree logic. Finding 1-2 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IE-C9 IE-C11 IE-D1	Finding 1-3: IE-D1 (MET), IE-C9 (MET), IE-C11 (Not MET) For the special initiator FT modeling, the following potential recoveries were modeled. Some bases have been provided: - For Loss of SW initiator FT, a repair BE was modeled with a probability of 1.0: BSWRXLOSTPUMPH-. It appears that this BE is risk significant as shown in the CDF results. Per staff, For loss of service water we do not plan on crediting repair until additional data becomes available. - For Loss of TBCCW & RBCCW initiator FT, several operator actions are modeled. BE 1RBSYLKGRCVR-H-- is assumed to have a probability of 0.5. Based on risk importance report, these events appear to be not significant. Per staff, the TBCCW and RBCCW recovery events are discussed in G.26 of the Component data. NRC Information Notice 98-25 details three examples of leaks from closed loop cooling water systems. The leak event at Palisades was isolated within 15 minutes and the system was restored to service. The other two events were minor and would have only resulted in adverse effects under specific conditions. Quad has no history of leak events and an estimation of 0.5 for recovery based on recovery on one of one event is considered conservative. However, this detail has not been documented in the Data Notebook - There are several recovery actions that appear to be important (see below, e.g., 2DCRX-BUS2RECF--, 1TBRXIERECVR-H--,). Per staff, the recovery of the DC bus is discussed in Section 3.5.5 of the IE notebook. The RHR recovery is discussed in Appendix G.36. The TBCCW recovery event is discussed in Appendix G.35. Most of these recoveries are conservatively assessed and the technical bases should have better plant-specific analyses. The current bases were not performed in a manner consistent with the applicable requirements in Human Reliability Analysis.	SR IE-C11 is Not Met. BSWRXLOSTPUMPH—is a basic event crediting repair of a SW pump failure and is conservatively modeled with a failure probability of 1.0. This is conservative relative to the ILRT application. 1DCRX-BUS1RECF-- and 2DCRX-BUS2REC F-- are typically in the same cutsets and have a FV of 0.13. The failure probability of recovering one of two DC buses is assigned 0.71. This is traceable to a failure probability of recoverying two busses (0.5) found in NUREG-0666. This number is considered conservative given Quad Cities has alternate 125 VDC batteries and chargers near the primary 125 VDC batteries and chargers. 1RBSYLKGRCVR-H-- and 1TBSYLKGRCVR-H—are leakage recovery events, with FV 6E-05 and 3.5E-4 respectively. The failure probability of 0.5 for each is considered conservative. 1TBOPIERECVR-H-- FAILURE TO RECOVER TB-IE INITIAL FAILURE IN MTTR WINDOW has a FV of less than 1.0E-4 and a failure probability of 0.5. No other equipment basic event recoveries are found in the 1E-12 truncation limit baseline cutset report. Based on the conservative failure probabilities for DC bus recoveries and low FV importance measures for all but the DC bus recoveries, the impact to the ILRT application given a refined approach is applied is considered negligible. Finding 1-3 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IE-C14	Finding 1-5: IE-C14 (MET CC-II) For ISLOCA, the IE frequency calculation shows that the value is 1.44E-7 /yr in Table 4-1 of the IE notebook. However, the Appendix A Table A.5-1 shows ISLOCA pipe rupture frequency of 1.69E-8/yr. Per staff, the ISLOCA value in Table A.5-1 is the correct value. Table 4-1 and the model probabilities need to be updated. Also about the ISLOCA modeling, the master fault tree does not show the failed systems/functions by %ISLOCA. ISLOCA frequency includes several unscreened lines, which should assume these affected systems failed due to the ISLOCA impact. Per staff, Section 10.2 of the Event Tree Notebook details ISLOCA sequences. For ISLOCA a 0.95 probability is assigned to CS failure due to the high likelihood of failure of the system due to the location of the break or the environmental conditions in the Reactor Building. LPCI operation is not credited for ISLOCA scenarios. For RHR a similar basic event with a probability of 0.95 is applied for failure of containment heat removal. (This F&O originated from SR IE-C14)	Meets Capability Category II or Above. Capability Category I is adequate for this application. ISLOCA frequency should be 1.69E-8 not 1.44E-7. The ISLOCA pipe rupture frequency used in the model is conservative. ISLOCA sequences do not impact the EPRI Class 3a and 3b frequencies. Finding 1-5 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
HR-I1 HR-D2 HR-D5	Finding 1-9: HR-I1 and HR-D5 are Met at CC I-III, HR-D2 is Met at CC-1 Significant pre-init HFEs have detailed analyses and been documented in Appendix A.2 of the HRA notebook. However, the documentation of the detailed analyses is not complete and some risk significant HFEs may need to be developed (or documented) according to the final quantification results. Example of potential documentation issues include: - 1HI-HPCI-TRA--HU: not documented in the App. A.2 - 1CAHU263-23--HCC: not in Section 4.5.4 (1CAHU263-52ABHCC is documented) - 1CAHU263-23--HCC: estimate is not consistent - 1CAHU263-52ABHCC: estimate is 8.00E-05 in RR database, but 4.95E-5 in HRA notebook - 1HI-HPCI-TRA--HU: detailed calculation appears to be not documented - 1RSMV1001-4B-VHU: detailed calculation appears to be not documented - 1RSMV1001185BVHU: detailed calculation appears to be not documented - BEPHU-1/2EDG-H--: detailed calculation appears to be not documented - BSSMV2-29011--H--: detailed calculation appears to be not documented	Capability Category I is MET Capability Category I is adequate for this application. Probability and FV for the documentation issue related BEs are shown below. Basic Event — Probability — FV 1CAHU263-23--HCC — 8.00E-05 — 0.00125 1CAHU263-52ABHCC — 8.00E-05 — 0.02839 1HI-HPCI-TRA--HU — 8.00E-04 — 0.0012 1RSMV1001-4B-VHU — 1.00E-04 — 0.00015 1RSMV1001185BVHU — 1.00E-04 — 0.00015 BEPHU-1/2EDG-H-- — 8.00E-03 — 0.01485 BSSMV2-29011--H-- — 8.00E-04 — 0.001 A review of the pre-initiators with issues identified by the peer reviewers found three with risk significant Fussell-Vesely values. 1CAHU263-52ABHCC has a probability of 8E-05. The HRA notebook documents the HEF at 4.95E-5. The error results in a higher CDF. The impact to this applicaiton is considered negligible and conservative. BEPHU-1/2EDG-H-- utilizes a detailed calculation applied to a similar action. The HEP of 8E-03 is developed under Calculation A.2.14 1EPHU-1-EDG-H--, *Preinit: Failure to Restore EDG 1 to Operability after Maintenance* is grouped with BEPHU-1/2EDG-H--. The HEP for 1EPHU-1-EDG-H-- is applied to BEPHU-1/2EDG-H--. This is documented in Appendix K, Section K.6 of the HRA notebook. Based on the above analysis, Finding 1-9 identified one error for a risk significant basic event (1CAHU263-52ABHCC). This error is conservative with regards to the ILRT application. Finding 1-9 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
AS-B3 AS-C1 AS-A7 QU-A1	Finding 1-11: AS-B3, AS-C1 and QU-A1 are Met at CC I-III; AS-A7 is Met at CC I-II A sample check of the QC ET's against the documented ET printouts in QC-PSA-002 Revision 2 as well as the sequences in the mater fault tree identified the following issues: 1. ET Notebook Figure 10.1-1, Large LOCA Outside Containment Event Tree, has two sequences with Class IIL, which are unnumbered and not included in the quantification (i.e., master FT). The system top “SPC-F” and “VTN-F” associated with these sequences are not modeled. Per staff, this BOC event tree has not been reflected in the final quantification accurately yet. Per staff, the BOC Event tree is incorrect and needs to add the two IIL sequences to the single top model. 2. Another sample check identified that the general transient ET is not consistent with the Fig 3-1 in ET Notebook. For example, GTR-033 & 035 are shown as CD sequence in Fig 3-1 but are OK sequence in the actual ET. Please note the above are only results from a small sampling check. (This F&O originated from SR AS-A7)	SRs met at Capability Category II or above. Capability Category I is adequate for this application. The BOC Event tree is incorrect and needs to add the two Class IIL sequences to the single top model. The MOR was modified to perform a sensitivity run. With the two class IL sequences added, CDF and LERF increased less than 1%. BOC failures do not directly impact the ILRT delta risk (Not included in calculation of EPRI class 3a and 3b). The missing BOC sequences were the result of not naming the missing sequences in the BOC event tree. All other event trees were reviewed and each event tree sequence was found to have been appropriately identified and included in the MOR. The GTR-033 & GTR-035 sequences are correct as OK sequences. The figures in the notebooks need to be updated. This is a documentation issue only. Finding 1-11 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
QU-C1	Fiinding 1-14: QU-C1 is not Met The seeding of the HFEs shown in the combinations appears to be done inconsistently, per discussion with the staff. A search of QC RR database showed that >300 HEPs applicable to IEAP quantification were not set to a seeding value such as 0.1. Examples of the inconsistent BEs are as follows: BSWOPB14-24SWH-- BSSOP-MAN-TR-H-- OP-ACT-BOP-RES (note the above BEs were identified from a small sampling check) (This F&O originated from SR QU-C1)	SR QU-C1 is Not Met at Capability Category I To identify potential combinations of concern, the HEPs in the RR database were increased to 0.1 (or maintained, if already above 0.1), and cutsets generated. Combinations from those cutsets were assessed for dependencies, and a recovery rules file created in order to replace independent HEPs with a joint HEP reflecting the levels of dependence assigned by the analysts. The HEPs in the RR database were then set back to their nominal values – and thus, when reviewed, did not reflect seed values. A flag file should have been created that would set the HEPs of all HFEs to 0.1 (or maintained, if already 0.1 or higher), and that flag file should have been used in the quantification process to ensure that the previously identified combinations re-appeared, so that the recovery rules file would properly address the joint HEPs of the combinations. Without that flag file, using just the nominal HEPs, some combinations fall below the truncation level used during quantification, the recovery rules are not applied to them, and therefore the CDF may be under-estimated. The resolution was to complete a sensitivity analysis, replacing nominal HEPs with the 0.1 values, to determine the extent of the impact. The CDF increased by less than 1%. Givne the low CDF risk increase, Finding 1-14 does not impact the conclusions of this application.

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2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IE-C15 IE-D3 QU-E3	Finding 1-16 IE-C15 and IE-D3 Met at CC I-III, QU-E3 Met at CC I-II One issue is identified during peer review is that the RR database does not have all error factors specified. Per discussion with the staff, this database was the one used for UNCERT run. Therefore, this could be part of the contributors to the low EF of the final CDF/LERF UNCERT runs. For example, %LOOP does not have EF specified, which is a significant contributor to final risks. (Also applies to %BOC-TB, others).	SRs met at Capability Category II or above. Capability Category I is adequate for this application. Specifying missing error factors would only impact the parametric uncertainty evaluation. It would not impact this application. Finding 1-16 does not impact this application.
AS-A9	Finding 1-17: SR AS-A9 is met at CC I-II The BE 1CS--CSLPIBOCF-- is modeled under LPI-BOC for FAILURE OF CS/LPCI FOR LOW PRESSURE INJECTION (BOC). Similar issue exists for BE 1CS--CSISLOCAF-- for ISLOCA. Per staff, The basic event 1CS-CSLPIBOCF-- is discussed on the bottom of page 10-4 and page 10-5 of the Event Tree notebook. There appears to be a typographical error with the event name. Moreover, detailed HRA should be performed for these events due to their risk significance.	SRs met at Capability Category II or above. Capability Category I is adequate for this application. 1CS--CSLPIBOCF-- and 1CS--CSISLOCAF-- are assigned probabilities of 0.5 and 0.95 respectively. The probabilities represent both phenomological estimates as well as operator action HFEs. Basic Event 1CS--CSLPIBOCF-- has a probability of 0.5 despite the fact that the pipe segment failures are all well above TAF. Both environmental effects and the ability of operator actions to control RPV level would be critical to the success. A detailed HRA would likely not significantly impact the basic event probability, given phenomological uncertainty. This event has a FV of 3.8E-4. BOC sequences are EPRI Class 8 Sequences where containment bypass occurs. Therefore, no impact to this application. Basic event 1CS--CSISLOCAF has a high failure probability. Likewise a detailed HRA would likely not make a significant impact to event probability. This event is associated with ISLOCA sequences. ISLOCA sequences are EPRI Class 8 Sequences where containment bypass occurs. Finding 1-17 does not impact the conclusions of this application.

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2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
LE-C10 LE-C12	Finding 1-18: LE-C10 Met at CC I Significant accident sequences were listed in Section 3.5 of QC-PSA-013 but were not reviewed to support equipment operation or operator actions during accident progression to reduce LERF, therefore, this SR is assessed Not MET CC-II. (This F&O originated from SR LE-C10)	Capability Category I is MET Capability Category I is adequate for this application. The top 10 LERF Sequences constituting 90% of total LERF were reviewed. The #1 Sequence contribution is Accident Class V (unisolated LOCA outside containment). Accident Class V sequences are EPRI Class 8 Sequences and have no impact to EPRI Class 3a or 3b. The #2 and #6 are accident Class IV (anticipated transient without scram (ATWS) These are EPRI Class 7 that typically would be LERF events, regardless of pre-existing liner leakage. The remaining LERF sequences would also have an insignificant impact to this application as the ILRT methodology is sensitive to changes in CDF, not LERF. For the above reasons, Finding 1-8 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
LE-C11	Finding 1-19: LE-C11 is Met at CC-I Reviewed: Level 2 notebook (QC-PSA-015, rev. 7), Sections C.13 and C.14; CET: QC1A. Table C.13-2 provides a summary of the quantification of Node NC in the CETs. Node NC queries whether an accident sequence includes a large containment failure. Structural evaluations are used to assign a probability to this event, for example 0.34 or 0.2 (see NC1 and NC2 in the table). Review of Section C.14, Long-Term Coolant Inventory Makeup (Node MU) indicates that SBCS is the only external water source considered in the MU mode and only those injection sources external to the reactor building are used in the MU node (due factors including loss of NPSH on primary containment blowdown). The operator action (1MUOP-ALT-RB-H--) and the recovery action (1MURX-ALIGN--H--) were noted – the operator action seems to be a Level 1 action (in HRA notebook) and the basis for the recovery action was not found. In addition, the failure of LPI due to NPSH after primary containment venting (node MUV1) is given a probability of 0.5. The basis for this probability is engineering judgment. The justification for these events was not sufficient.	Met at Capability Category I Capability Category I is adequate for this application. This finding is judged to be primarily a documentation issue. 1MURX-ALIGN—H-- with a failure probability of 3.9E-01, is used to model recovery of SSMP failure and SBCS. 1MUOP-ALT-RB-H-- with a failure probability of 4.20E-01 is applied to the operator action to switch injection to alternate systems ouside the Reactor Building. These basic events are included in Level 2 logic only. The BE LERF FV is 1E-03. Therefore, these operator actions have a negligible impact to this application. The failure of LPI due to NPSH after primary containment venting is represented by basic event 1MUPH-NPSH---F-- *NPSH CAUSES FAILURE* and has a probability of 0.5. This basic event is not included in Level 1 or Level 2 cutsets above 1E-12/yr truncation limit. Therefore, no impact to this application. Overall, Finding 1-19 has a negligible impact to CDF. Finding 1-19 does not impact the conclusions of this application.
DA-D1	Finding 2-1: DA-D1 is Met at CC-I Entry SB DG X (SBO DG fails to run) in Table C.2-2 of the Component Data notebook (QC-PSA-010, rev. 7) indicates a run time of 204 hrs. But the notebook on p. F-14 gives a total of 162+162=324 hrs. In a discussion with the staff, it appears that the data on p. F-14 are in error and that the correct data are those given in Table C.2-2. Thus, a preliminary conclusion is that this would be a documentation issue.	Met at Capability Category I Capability Category I is adequate for this application. The calculation used 204 hours. A review of input data confirmed that 204 hours is correct. Therefore, finding 2-1 is a documentation issue only. Finding 2-1 does not impact this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
DA-C1	Finding 2-4: DA-C1 is Met at CC I-III Section G.36 of the Component Data notebook (Volume 1) addresses the recovery of RHR (repair). It is in the model under Basic Event 1RHOPREPAIRTRH-- and is based on generic mean time to repair data from WASH-1400 and IEEE 500. However, it appears that there is no discussion that these generic repair data are consistent with the test and maintenance philosophies of the plant. The staff provided additional plant-specific RHR repair data, which appear to be compatible with the generic repair information.	Met at Capability Category I-III Capability Category I is adequate for this application. Because of the extended time available and the lack of radiation in the work environment, the time window over which repair can be effectively carried out can be quite long (e.g., 20-33 hours for Class IIA and Class IIL). WC-AA-104 Integrated Risk Management Revision 24 notes that if a work activity needs to be worked immediately, the Shift Manager may perform the risk screening and present the results to the Station Duty Manager. The current process assists to expedite the repair work. QC current maintenance practices include staffing 24 hours/day during the work week. Maintenance personnel are available within 2 hours during weekends. Operation personnel have the capability to install pipe clamps to address pipe leakage.[1] Given the long duration, expedited screening process and maintenance department staffing, the generic repair data is consistent with the test and maintenance philosophies of the plant. Finding 2-4 does not impact this application.

[1] Staffing information and Operations capability obtained from the QC Maintenance Department Director direction via email on 3/28/17.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
DA-D1	Finding 2-5: DA-D1 is Met at CC I Section C.2.3 in Appendix C of the Component Data notebook (QC-PSA-010, rev. 7) indicates three conditions for a component to be included in the plant specific data analysis: 1) the component is included in the PRA, 2) the component has the potential to an important contributor to CDF or LERF, and 3) the component is included in the Maintenance Rule program. Because the above approach does not mandate a post-quantification review of data, there is a possibility of basic events that are significant in terms of risk, might not have been included in a plant-specific analysis, and therefore does not meet Capability Category II of SR DA-D1. This was confirmed through a sample review of significant basic events, which are listed in Table F-1 of the PRA Quantification notebook (QC-PRA-014, rev. 3). The sample review included the basic events with Fussell-Vesely values greater than 1.50E-02. The review showed that two basic events, 1ACCB1412----D-- and 1ACCB1427----K--, representing the failure of breakers to open and close, and whose Type Code is CB F, were not updated with plant-specific data. (This F&O originated from SR DA-D1)	Met at Capability Category I Capability Category I is adequate for this application. The two basic events are 4 KV breaker events with a generic failure rate of 2.70E-3. Use of generic data meets Category I. A sensitivity was performed by doubling the failure probability of basic events 1ACCB1412----D-- and 1ACCB1427----K-- in the base CDF cutset. CDF increased by 4.1%. CDF increased by 4.1%. Increasing the failure probability would not cause a significant change that would be considered significant given the current ILRT risk metric margins. Finding 2-5 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
HR-I1 HR-G4	Finding 2-7: HR-I1 is Met at CC I-III, HR-G4 is Met at CC I-II The action to emergency depressurize (1ADOP-DEP-ADSH--) was reviewed. The justification for the derivation of the timing data and dependency level could be strengthened. Namely: 1) Strengthen the basis for the time window of 40 min. In the HRA Calculator, the time window is given as based on data from MAAP run QC05014A (which shows a time to core damage of 32 min in absence of injection and 5 SRVs opened at 15 min), and from MAAP run QC05016 (which shows that depressurization with 2 SRVs opened at 35 min leads to LPCI injection at 49 min and no core damage). From these two MAAP runs, the derivation that the 40 min is appropriate does not appear to be clearly established. It seems that other MAAP runs (e.g., QC05014) could be added to establish the 40 min time window. 2) Strengthen the justification for using a low dependency level for the execution of the action. Because by procedure the operators should not emergency depressurize until TAF (-142 inches) is reached, and TAF is reached at 27 min based on MAAP run QC05014, there are only 40-27=13 min for the execution of the action. This time is associated with a moderate dependency in the HRA Calculator method. Selecting a low dependency level may be appropriate for the execution of the action, but a justification is needed. Note that this comment about the dependency level only applies to the execution portion. It does not apply to the cognitive portion, which is fine as-is. 3) There are other similar HFEs that use the same 40 min time window and for which the justification needs to be more firmly established.	SRs met at Capability Category II or above. Capability Category I is adequate for this application. 1) A review of MAAP cases confirmed that 40 minutes is a reasonable for the time window. MAAP case QC05001 is a case with MSIVs closed, no injection and no manual RPV Emergency Depressurization. Core Damage occurs at 54 minutes. 2) Using procedure QCA-100, operators will attempt to control RPV water level between 0" and 48". Through training and this procedure, the operators are well aware of the potential need to depressurize; the 0" level is their first (and anticipatory) cue. They will not wait until -142" to begin preparing for depressurization – they will be anticipating the potential for that action, starting when water level reaches 0". Thus, the delay time (2 minutes) used in the development of the HEP for this HFE is appropriate; the amount of time available for recognizing the need for the action and preparing for the action is approximately 38 minutes. Control room resources (e.g., shift supervisor, STA) will be focused on the procedure steps, and will have immediate feedback if the reactor operators fail to take the necessary depressurization steps at the appropriate times. Thus, a level of low dependence is appropriate for the ability to recover if an error of omission is made during the performance of the depressurization action. 3) As noted above, 40 minutes is reasonable based on MAAP case QC05001. Finding 2-7 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
QU-A5 QU-C1 QU-C2 HR-D5 HR-G7	Finding 2-8: QU-A5 and QU-C2 and HR-D5 are Met at CC I-III, QU-C1 and HR-G7 are not Met at CC I The HRA uses two approaches to evaluate HFE dependencies. However, it appears that overall not all relevant HFE combinations were evaluated. For example, a cutset with a CDF of 1.05E-08/yr involves HFEs: BSSOP-DUALIEXH--, 1ADOP-DEP-ADSH--, and BACOPSWTCHBUSH--. While 1ADOP-DEP-ADSH-- and BACOPSWTCHBUSH-- appear to have been analyzed for dependencies and are replaced with dependent group 1--RX-HPI-009-H-- in the cutset, the dependency with BSSOP-DUALIEXH-- is not captured. Note that according to Section 5.2.3.3 of the HRA notebook, HFEs with an HEP equal or greater than 0.5 have not been analyzed for dependencies (based on the consideration that such high HEPs capture potential dependencies), but none of the above HFEs fall into that category. Another example is a cutset with a CDF of 7.91E-09/yr, where a dependency group has been considered (1--RX-SPCE-011-H--) but it does not appear to have included HFE BSSOPEBYP-TR-H--. More generally: 1) The method of using basic events directly inserted into the fault tree to model dependent events makes the review of cutsets difficult. For example, some of the cutsets will have the HFEs that are part of a given combination. To ensure that the dependency has been correctly captured in the results, the reviewer has to make a search of the cutset files to identify the analog cutset with the same random failures and the dependent HFE. This makes the process cumbersome. 2) The HRA Calculator used with the dependency analysis file provides a transparent and traceable treatment of HFE combinations. It provides more data than the summary information given in Table 5.4.2-1.	SRs HR-G7 and QU-C1 Not Met at Capability Category I. For the example BSSOP-DUALIEXH—(FAILURE TO SELECT UNIT AS THE PRIMARY INJECTION PATH) – this HFE pertains to the action of swapping a Safe Shutdown Motor Pump (SSMP) between units. Its nominal HEP is 0.46, which is very close to the 0.5 threshold at which point the event would not have been considered for dependencies anyway. Thus, the impact is judged to be minimal. For the example of BSSOPEBYP-TR-H (Operator Manually Closes SSMP Room Cooler Bypass (Early) (Transient)) – this clearly meets the definition of a “SSMP room cooling HEP”. The HRA Notebook, in Section 5.4.2, explains, “…SSMP Room cooling HEPs were eliminated because these are addressed by the intervening success of the SSMP initiation actions. If SSMP initiation fails, SSMP is failed and room cooling actions are not.” Here, “eliminated” means removed from the HRA database used during the identification of combinations. Therefore, BSSOPEBYP-TR-H-- was not considered for combinations. The examples provided do not demonstrate that dependencies were inappropriately treated. To identify potential combinations of concern, the HEPs in the RR database were increased to 0.1 (or maintained, if already above 0.1), and cutsets generated. Combinations from those cutsets were assessed for dependencies, and a recovery rules file created in order to replace independent HEPs with a joint HEP reflecting the levels of dependence assigned by the analysts.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
Finding 2-8 continued	Finding 2-8 contiued.	Item 2 (HRA Calculator) is considered a recommendation for a future update. Finding 2-8 does not impact the conclusions of this application.
HR-I1 HR-G6	Finding 2-9 HR-I1 is Met at CC I-III, HR-G6 is not Met While it was indicated during a discussion with the PRA analysts that a reasonableness check was done by comparison to previous values, there is no documentation of the results. The SR calls for a verification of the HFEs relative to each other to check their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	HR-G6 is not Met at CC I A reasonableness check table was developed and reviewed but not included in the 2014 FPIE PRA update documentation. The review has been re-verified and will be included as part of the next FPIE PRA update. Note, the reasonableness check is done given the plant history, procedures, operational practices, and experience. HRA Notebook Section 3.13 Operational Trends documents a review of plant experience with regard to operator actions modeled in the PRA. QC human performance monitoring system results were reviewed. These included trends in Operational Focus, Industrial Safety, ERO/TSC performance and INPO Human Performance Event Rate trends. The HRA conclusion noted that “the human interface with the safe operation of Quad Cities is above average in the industry and no decrement to the nominal HEPs developed here is needed.” This is a documentation issue only. Finding 2-9 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
QU-B3	Finding 2-10: QU-B3 is not Met Section 3.7 of the Summary Document notebook (QC-PRA-013, rev. 7) documents the convergence evaluation performed for the PRA. Regarding LERF, a truncation limit of 1E-13/yr reads to an increase in LERF of 7.3% compared to a truncation limit of 1E-12/yr. This is more than the 5% cited in the SR. In addition, Figure 3.7-2 which plots the LERF for various level of truncation limits does not show clear convergence. Therefore, it appears that convergence for LERF has not been firmly established.	QU-B3 is not Met To determine the impact to this application, the FPIE PRA model was quantified at 2E-14/yr truncation limit. Convergence is shown from the 2E-13/yr truncation limit. The respective LERF values are 2.19E-7/yr at 2E-14/yr and 2.10E-7/yr at 2E-13/yr. The increase in LERF is 4.2% which is below the 5% cited in the SR. If LERF results of 2.10E-7/yr at a truncation limit 2E-13/yr were used instead of 1.97E-7/yr at a truncation limit of 1E-12/yr, the impact to this application would be be to total LERF only. There would be a small change to total LERF and the significant margin to the LERF threshold would remain in place. Finding 2-10 does not impact the conclusions of this application.
QU-D5	Fiinding 2-11: QU-D5 is Met at CC I-III QU-D5 requires that a sampling of nonsignificant cutsets or sequences be performed to determine that they are reasonable and have physical meaning. While the Quantification Notebook (QC-PSA-014, rev. 3) indicates that this was verified, there is no details of that review. The quantification should provide a list of nonsignificant cutsets or sequences reviewed and a justification that they are reasonable and have physical meaning. (This F&O originated from SR QU-D5)	QU-D5 is Met at CC I-III Capability Category I is adequate for this application. A review of significant and nonsignificant cutsets and accident sequences was performed for the 2014A PRA model update in accordance with the 2009 ASME/ANS PRA Standard RA-Sa-2009. This review occurs through out model development and also takes place during an independent challenge review of the PRA results prior to final approval. SR QU-D5 Basis for Assessment stated “the documentation of the review of non-dominant cutsets is brief and requires additional detail.” This is documentation only. Finding 2-11 does not impact this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IE-A6 IE-D1 IE-D2	Finding 3-2: IE-A6 is Met at CC I-II, IE-D1 is Met at CC I-III, IE-D2 is not Met QC-PSA-001 Initiating Events The IE notebook does not explicitly address common cause failure or maintenance for initiating events in the special initiating events review of plant systems. QC-PSA-005 Systems NB The Systems Notebooks Section 6.1.7 Addresses CCF of equipment in the systems, however it is not clearly addressed whether the CCF would lead to an initiating event and are being identified and fed back to be the initiating events analysis. QC114A-CDF-LERF.caf Review of the QC Model Fault Trees shows that potential common cause and alignment are generally addressed in the fault trees supporting initiating events, however the effects of maintenance are not (1.200 clarification). Examples: SW-IE no maintenance events included RBCCW-IE no maintenance events included TBCCW-IE no maintenance events included TBCCW-IE – No HX CCF – for example 1TBHE3802A/B-PCC TBCCW-IE – No Pump CCF – for example 1TBPM3801A/B-XCC In some cases, Tech Spec allowed configurations which allow one train to be inoperable for a period of time are credited for screening (ex 2.3.3.16 RHR SW). For Some systems (2.3.3.13 HPCI, 2.3.3.14 RCIC) one system out of service is allowed per Tech Specs given the alternate mitigation strategy is available. However a common cause failure of both strategies or failure of one strategy with the other under maintenance would trigger a unit shutdown. The basis for screening the initiator under these circumstances is not discussed in the systems initiator review however, the staff was able to show that cases of rapid Tech Spec shutdown are subsumed in the Manual Shutdown initiating event.	IE-A6 is Met at CC I-II, IE-D2 is not Met (IE-D2 is documentation SR). IE-A6 requires including common cause and is met at CC II which is adequate for this application. Although the IE noebook does not explicitely address common cause failure or maintenance criteria for initiating events review of plant systems, the potential for these types of failure was not excluded. This is apparent in the selection of system initiating events for modeling. Systems with the potential for special initiators include systems that typically would require multiple train failures. These systems include TBCCW, RBCCW, Circ Water and Loss of Instrument Air Systems. Regarding the effects of maintenance on fault tree supporting initiating events: The SW-IE maintenance unavailability is assigned to the swing (unit ½) pump. Basic event BSWPM1/2-3901M-- has a probability of 6.77E-02. The RBCCW maintenance unavailability is assigned to the RBCCW swing (Unit ½) pump. Basic event BRBPM1/2-3701M-- has a probability 3.57E-03. TBCCW IE maintenance unavailalability is assigned to 1TBPM1-3801A-M-- and 1TBPM1-3801B-M--, each with a failure probability of 4.11E-03. As noted, the failure of multiple trains of mitigation systems, such as HPCI/RCIC or RHR may lead to rapid shutdowns, however these events are shown to be subsumed in the Manual Shutdown initiating event. Finding 3-2 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IE-C12 IE-D2	Finding 3-8 IE-C12 and IE-D2 are not Met QC-PSA-001 IE The roadmap provided points to Appendix G as a comparison of the initiating events back to generic sources. Appendix G develops an initiator frequency for LOFW due to motor driven FW pumps since it is noted that the NUREG 6928 data is dominated by Turbine Driven pump failures. Discussion with the staff is that no comparison to generic data sources is available. (This F&O originated from SR IE-C12)	IE-C12 and IE-D2 are not Met (IE-D2 is a documentation SR). %TF has a initiating event frequency of 2.90E-02 in the QC FPIE PRA model. The %TF Birnbaum is 1.53E-06. This low Birnbaum importance is attributal to QC configuration with multiple high pressure injection sources. Safe Shutdown Make-up Pump SSMP, HPCI and RCIC pumps provide multiple and diverse high pressure injection sources. An upper bound would be to use NUREG CR-6928 2010 Loss of Feedwater data mean frequency of 6.89E-02. This would result in a CDF increase of 2%. This CDF increase would not result in a change to the ILRT conclusions. As noted above, the Loss of FW initiating event has a low importance and there is a higher reliability of FW motor driven pumps compared to FW steam driven pumps. Finding 3-8 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
DA-C3 IE-C2 IE-C7 IE-D1	Finding 3-9: DA-C3 and IE-C2 are Not Met, IE-C7 and IE-D1 are Met at CC I-III The data selected for initiating event frequencies generally spans only 4 years, Initiating Events Analysis Section 3.2 bases this on improved performance, however review of Tables B1 & C1 indicates that including data since 2003 would double the initiating events to be considered (not including manual scrams for outages). Discussion with the staff is that older initiating events are included in the industry data that is used for the generic prior. However, this conclusion is not validated and no evidence is provided via design or operational change that the Initiating events from the plant historical data are no longer applicable. This approach discounts actual events that occurred earlier in Station history, such as a LOOP occurred on QC unit 2 from power operation on Aug 2, 2001, due to a lightning strike. The same issues are identified in data analysis when plant-specific data collection only spans for 4 years. The basis does not discuss the design or operation changes to support exclusion of plant specific data history. (This F&O originated from SR IE-C2 and DA-C3)	DA-C3 and IE-C2 are Not Met The peer review did not find sufficient evidence to warrant exclusion of initiating events prior to the data collection period that began on 1/1/2010. The Quad Cities initiating events for anticipated transients are derived using a Bayesian update of generic data sources. Therefore, the anticipated transient frequencies are known to be consistent with the generic evidence. A review of the 10 year period from Jan. 1, 2007 to Dec. 31, 2016, found only 4 general transient (2 Turbine Trips, 1 Loss of Condenser and 1 MSIV Closure). This equates to ~0.2 trips per unit per year. Incorporating a longer data collection period, would have a negligible impact on CDF. Regarding LOOP data, the Initiating Event Notebook contains data going back to 1985. In the approximately 60 years (2 units for ~30 years) of operating experience, there has been only one LOOP event. Incorporating the LOOP event of 2001, would have a negligible impact on CDF. Given the above, Finding 3-9 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
HR-E3 HR-E4	Finding 3-15: HR-E3 and HR-E4 are Met at CC I-III In reviewing selected HEPs in Appendix A.1, in some cases conservative assumptions are made in the HEP development due to a lack of 'talk-though' of scenarios (1ADOPSPURFUSEH--, 1ADOP-TRC-ADSH--,1ADOP-TRC-ADSHCC, others). If these HEPs appear in significant cutsets, operator talk throughs should be performed as with the other significant HEPs. (This F&O originated from SR HR-E3)	Meets Capability Category II or Above. Capability Category I is adequate for this application. The refinement of these HEPs may lower CDF by removing conservatisms. For example the HRA calculation for 1ADOP-TRC-ADSH-- states: "ASEP nominal is assigned. Once a talk-through is available, it is likely that ASEP "Lower Bound" could be assigned. This (talk-through) would also need to verify that operators practice the event iin the simulator." Since the ILRT application is driven by CDF sequence contributions, the impact of this finding is conservative with regards to ILRT results. Finding 3-15 does not impact the conclusions of this application.
SY-C2 SY-A8	Finding 3-20: SY-C2 and SY-A8 are Met at CC I-III Looking at Equipment boundaries modeled vs. component boundaries in the Data Analysis QC-PSA-010 Data Vol 1 Table B-1 DG failure rate includes output breaker QC-114A-CDF-LERF.caf EDG models the output breaker separately: - as part of the power output circuit logic 1ACCB1429----F-- - as part of the hardware logic 1ACCB1429----M-- Discussion with the staff is that this appears to be redundant modeling as the output breaker is already included in the data for EDG failure. (This F&O originated from SR SY-A8) This does not appear to be a systematic issue. However, this F&O is a finding because both BE's included in the F&O description are risk-significant.	Meets Capability Category II or Above. Capability Category I is adequate for this application. Removal of the EDG output breakers from the PRA model to eliminate the redundant modeling would reduce CDF. Since the ILRT application is driven by CDF sequence contributions, the impact of this finding is conservative with regards to ILRT results. Finding 3-20 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
IE-A5 IE-A8 SY-C2 SY-A4 SY-B8 SY-C1	Finding 3-21: IE A5, IE-A8 are Met a CC I-II, SY-C2, SY-B8 and SY-C1 are Met at CC I-III, SY-A4 is not Met Reviewed: QC-PSA-005 System NB, Appendix E The system Manager interview is credited in various aspects of the development and confirmation of the quality of the PRA model. Per the System notebooks, No system manager interview has been performed for: -Common Actuation system -Drywell Cooler System -On-Site water Systems -Primary Containment Isolation System -Reactor Protection System -Recirc Pump Trip and Alternate Rod Insertion System -Vapor Suppression System (This F&O originated from SR SY-A4)	SY-A4 is not Met Interviews for all systems were not able to be completed in the 2014A update based on plant personnel availability. These will be addressed in the next PRA update. It should be noted that the SRME interfaces with System Managers through System Health and Maintenance Rule interactions. The SRME also evaluates design changes with the potential to impact the PRA model. Given the SRM interactions, the omission of select interviews is judged to have a negligible impact on the PRA model. Finding 3-21 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
SY-A4 SY-B8 SY-C1 SY-C2	Finding 3-22: SY-C2, SY-B8 and SY-C1 are Met at CC I-III, SY-A4 is not Met Credit is taken for walkdowns performed as part of the internal flooding analysis in Section 2.5 of the Systems Analysis to address spatial dependencies. As documented in appendix A of the Flood notebook QC-PSA-012, these walkdowns were focused on flood-specific spatial aspects of water sources, targets, propagation pathways, spray, etc. No other walkdowns are documented to have been performed in support of confirming the systems analysis correctly reflects the as-built as-operated plant.	SY-A4 is not Met The SRME interfaces with System Managers through System Health, and Maintenance Rule interactions and also evaluates design changes with the potential to impact the PRA model. In addition there were extensive walkdowns of plant equipment performed in support of a Fire PRA update. The FPIE PRA update is the initial starting point for the Fire PRA. Plant changes impacting the Fire PRA are evaluated for impacts to the FPIE PRA. Plant modifications and procedure changes that potentially impact the FPIE PRA are entered into the “URE” database and addressed in the PRA update process. Due to the ongoing SRME interfaces, the plant modification and procedure change process, and Fire PRA walkdowns, the omission of FPIE specific walkdowns is judged to have a negligible impact on the QC PRA. Finding 3-22 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
DA-D4	Finding 4-3: SR DA-D4 is Met at CC I SR DA-D4 requires: When the Bayesian approach is used to derive a distribution and mean value of a parameter, CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. The Bayesian approach was used to update generic data. A reasonableness check of results (several methods are provided in the PRA Standard) was not found. (This F&O originated from SR DA-D4)	The SR is Met at CC I. Capability Category I is adequate for this application. As noted in the peer review report, "Component Data Notebook was reviewed. Tables 1-2 and 1-3 provide a check between the 2010 data and 2014 data for select typically risk significant equipment. However, no reasonableness check of the posterior distribution was found." Although not documented, the qualified PRA analyst would perform a reasonableness check during the documentation process. A reasonableness check was performed on risk significant component data as identified in the Component Data notebook Tables 1-2 and 1-3. Changes in probabilities were reasonable given comparisons to 2010 probabilities and generic data. Finding 4-3 does not impact this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
DA-E2 DA-C4	Finding 6-1: DA-E2 is Met at CC I-III, DA-C4 is Not Met The staff has communicated that maintenance rule functional failures were not screened against the PRA success criteria to determine applicability for the 2014 PRA data update. (This F&O originated from SR DA-C4)	DA-C4 is Not Met Maintenance Rule (MR) functional failure definitions are captured in MR T&RM ER-AA-310-1004 Maintenance Rule – Performance Monitoring. Detailed guidance is provided to assure functional failures are properly categorized by the System Manager. The MR technical guidance conforms to what is needed to identify equipment function failures within the PRA. The System Manager provides functional failure data to the risk management engineers. A qualified risk management engineer screens the functional failures to determine if the PRA model is impacted. For example, a Core Spray minimum flow valve failure to open is reported by the System Manager. The PRA system success criteria require the min flow valve to open and therefore, the failure is screened by the risk management engineer as impacting the clean water MOV fail to open (FTO) type code MV D. This process constitutes equivalence to screening against PRA success criteria. Finding 6-1 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
AS-C3 HR-I3 LE-F3 SC-C3 SY-C3 QU-E2 QU-E4 QU-F4	Finding 6-6: AS-C3, HR-I3, LE-F3 SC-C3, SY-C3 and QU-F4 are Met at CC I-III, SRs QU-E2 and QU-E4 are Not Met. As shown in the reviewed PRA notebooks, significant efforts have been dedicated to the identification of assumptions made in the development of the PRA model. However, the identification process has not been performed consistently in each PRA technical element. For example, system analysis notebooks have summarized and documented the assumptions while assumptions are still scattered in some other notebooks. Furthermore, it is expected all these assumptions be summarized and tracked in the summary document (or in a separate document). The treatment of these assumptions should be documented, especially the links to the existing sensitivity studies should be documented although the identification of key assumptions may be deferred to application-specific studies. For example, the Event Tree Notebook, PRA Summary Notebook (QC-PSA-013), has been reviewed. While some key assumptions are identified in the assumptions sections in the several appendices of the event tree notebook, and the associated sources of uncertainty are captured in the summary notebook, there are a large number of assumptions dispersed throughout the event tree notebook that are not captured. A word search of the event tree notebook reveals the word 'assumption' or 'assumed' is used a total of 194 times. However, the assumptions section only lists 11 assumptions, and the summary notebook lists 7 sources of uncertainty. (This F&O originated from SR AS-C3)	SRs QU-E2 and QU-E4 are Not Met. The assumptions made in the development of the PRA model are documented in the respective PRA notebooks. They are not summarized and tracked in the Summary document. However, the sources of model uncertainly and related assumptions are documented in Appendix B of the QC Summary Notebook based on the guidance provided in EPRI 1016737 (as endorsed in NUREG-1855). This assessment did address the items to consider per the EPRI guidance and indicated the potential sources of model uncertainty. This is a documentation issue only. Finding 6-6 does not impact this application.
SY-B1	Finding 6-9: SY-B1 is Met at CC I-III Reviewed: System Notebooks (QC-PSA-005) -Section 6 Component Data Notebook (QC-PSA-010) – Volume 2 While reviewing the instrument air modeling, it was discovered that although 1/3 air compressors are required for success at gate IA002, the CCF of 2/2 similar (out of the three total) air compressors is sufficient to fail the gate. (This F&O originated from SR SY-B1)	The SR is Met at CC I-III. Capability Category I is adequate for this application. The model correction from CCF of 2/2 to a CCF of 3/3 air compressors would provide a small reduction in CDF. Since the ILRT application is driven by CDF sequences, the impact of this finding is conservative with regards to ILRT results. Finding 6-9 does not impact the conclusions of this application.

TABLE A-2

2017 FPIE PEER REVIEW FINDINGS AND IMPACT TO APPLICATION

APPLICABLE ASME SR(S)	FINDING	IMPACT TO APPLICATION
SY-B3 SY-C2	Fidning 6-10: SR SY-B3 and SY-C2 are both Met at CC I-III Check valves were not modeled for common cause. This approach is consistent with the criteria for qualitative screening, established in the component data notebook, but does not appear to be justified by the industry guidance used, NUREG/CR-5497. Data for check valves is included for various failure modes in the NUREG and there is no discussion related to excluding them. (This F&O originated from SR SY-B3)	Meets Capability Category II or Above. Capability Category I is adequate for this application. Omission of CCF modeling of check valves is estimated to have a negligible impact on the PRA results and therefore on this application. Finding 6-10 does not impact the conclusions of this application.
SY-A15 SY-C1	Finding 6-14: SY-A15 and SY-C1 are Met at CC I-III Reviewed: -System Notebooks (QC-PSA-005) Components are grouped together based on a variety of data sources and the potential failure modes from SY-A14 are addressed as appropriate for the groups. Where components are screened based on SY-A15, it is noted in the table. However, the system notebooks' criteria differ from SY-A15. SY-A15 only lists two criteria, while the system notebooks list four different criteria. It is not clear from looking at the table if the correct criteria from the actual SR were used for screening. (This F&O originated from SR SY-A15)	Meets Capability Category II or Above. Capability Category I is adequate for this application. Additional criteria for screening as found in system notebooks is the following: 1) The screened contributors are position faults for components (such as those that occur during or following test and maintenance activities) for which the component receives an automatic signal to place it in its required state. 2) It can be shown that the omission of the contributor does not have a significant impact on the results. Criterian 1 above is seldom used. Failure to realign would impact the Fire PRA model. As the FPIE PRA model is used as input to the Fire PRA, this failure mode is typically modeled. Critieran 2 above is related to SY-A15 criteria (a) and (b) and would not change groupings. Removing the additional criteria would have a negligible impact on CDF. Finding 6-14 does not impact the conclusions of this application.

A.2.6 External Events

Although EPRI report 1018243 [A.8] recommends a quantitative assessment of the contribution of external events (for example, fire and seismic) where a model of sufficient quality exists, it also recognizes that the external events assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval. Based on this, currently available information for external events models was referenced, and a multiplier was applied to the internal events results based on the available external events information. This is further discussed in Section 5.7 of the risk assessment.

A.2.7 PRA Quality Summary

Based on the above, the QC PRA is of sufficient quality and scope for this application. The modeling is detailed; including a comprehensive set of initiating events (transients, LOCAs, and support system failures) including internal flood, system modeling, human reliability analysis and common cause evaluations. The QC PRA technical capability evaluations and the maintenance and update processes described above provide a robust basis for concluding that these PRA models are suitable for use in the risk-informed process used for this application.

A.3 IDENTIFICATION OF KEY ASSUMPTIONS

The methodology employed in this risk assessment followed the EPRI guidance as previously approved by the NRC. The analysis included the incorporation of several sensitivity studies and factored in the potential impacts from external events in a bounding fashion. None of the sensitivity studies or bounding analysis indicated any source of uncertainty or modeling assumption that would have resulted in exceeding the acceptance guidelines. Since the accepted process utilizes a bounding analysis approach which is mostly driven by that CDF contribution which does not already lead to LERF, there are no identified key assumptions or sources of uncertainty for this application (i.e. those which would change the conclusions from the risk assessment results presented here).

A.4 SUMMARY

A PRA technical adequacy evaluation was performed consistent with the requirements of RG-1.200, Revision 2. This evaluation combined with the details of the results of this analysis demonstrates with reasonable assurance that the proposed extension to the ILRT interval for QCNPS Unit 1 and Unit 2 to fifteen years satisfies the risk acceptance guidelines in RG 1.174.

A.5 REFERENCES

[A.1] Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk Informed Activities*, Revision 2, March 2009.

[A.2] Boiling Water Reactors Owners' Group, *BWROG PSA Peer Review Certification Implementation Guidelines*, Revision 3, January 1997.

[A.3] American Society of Mechanical Engineers, *Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME RA-S-2002, New York, New York, April 2002.

[A.4] U.S. Nuclear Regulatory Commission, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Draft Regulatory Guide DG-1122, November 2002.

[A.5] Reactor Oversight Program MSPI Bases Document, Quad Cities Generating Station, Revision 5b, December 15, 2011.

[A.6] ASME/American Nuclear Society, *Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME/ANS RA-Sa-2009, March 2009.

[A.7] *Quad Cities Nuclear Power Station PRA Peer Review Report*, BWROG Final Report, February 2000.

[A.8] Electric Power Research Institute (EPRI), *Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325*. EPRI TR-1018243, October 2008.

[A.9] Quad Cities Nuclear Power Station PRA Peer Review Report (Internal Flooding) Using ASME PRA Standards, November, 2010.

[A.10] QC-PSA-16 Self-Assessment of the Quad Cities PRA Against the Combined ASME/ANS PRA Standard Requirements Revision 2.

[A.11] QC-PSA-16, Self-Assessment of the Quad Cities PRA Against the Combined ASME/ANS PRA Standard Requirements Revision 5, January 2015.

[A.12] Probabilistic Risk Assessment (PRA) Peer Review Process Guidance (NEI 00-02) Rev. A3, dated March 20, 2000

[A.13] ER-AA-600-1014, Risk Management Configuration Control, Rev. 7

[A.14] ER-AA-600-1023, FPIE PRA Model Capabilities, Rev. 06

[A.15] *Quad Cities Nuclear Generation Station PRA Peer Review Report* (All applicable SRs except Internal Flooding), April, 2017.

[A.16] NEI 05-04, *Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard*, Revision 2, November 2008.

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