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10 CFR 50.90

RS-03-044

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

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Request for Amendment to Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program"

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The proposed change revises Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A test to no later than July 22, 2009, for Unit 1, and no later than May 16, 2008, for Unit 2.

TS Section 5.5.12 provides the requirements for the Primary Containment Leakage Rate Testing Program. TS 5.5.12.a requires that this program 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. Additionally, the testing is required to conform to the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995.

A plant-specific, risk-based evaluation has been performed in support of the one-time deferral of the Type A test frequency from once in 10 years to once in 15 years. The evaluation demonstrates that a change in the Type A test interval from 10 years to 15 years represents a very small impact on risk, as defined by NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, dated November 2002.

This request is subdivided as follows.

- Attachment 1 is the notarized affidavit.

- Attachment 2 provides an evaluation supporting the proposed change.

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- Attachment 3 contains the marked-up TS page with the proposed change indicated.
- Attachment 4 provides retyped TS pages with the proposed change incorporated.
- Attachment 5 provides the risk assessment supporting the proposed change.

The proposed changes have been reviewed by the QCNPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

The next QCNPS refuel outage is currently scheduled to commence on February 24, 2004. To support incorporation of the Type A testing changes into the schedule for the upcoming QCNPS Unit 2 refuel outage (Q2R17), EGC requests approval of the proposed amendments by February 1, 2004. Once approved, the amendments shall be implemented within 60 days. This implementation period will provide adequate time for station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91(b), EGC is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

If you have any questions or require additional information, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

Respectfully,

Patrick R Simpson

Patrick R. Simpson
Manager – Licensing
Mid-West Regional Operating Group

Attachments:
Attachment 1: Affidavit
Attachment 2: Evaluation of Proposed Change
Attachment 3: Markup of Proposed Technical Specifications Page Change
Attachment 4: Retyped Technical Specifications Pages for Proposed Change
Attachment 5: ERIN Report No. C46702044-5163, "Quad Cities Risk Assessment to Support ILRT (Type A) Interval Extension Request," December 2002

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

ATTACHMENT 1
Affidavit

STATE OF ILLINOIS	**)**	
COUNTY OF DUPAGE	**)**	
IN THE MATTER OF	**)**	
EXELON GENERATION COMPANY, LLC	**)**	**Docket Numbers**
QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2	**)**	**50-254 and 50-265**

SUBJECT: **Request for Amendment to Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program"**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

Patrick R. Simpson
Patrick R. Simpson
Manager – Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 27th day of

February, 2003.

Timothy A. Byam
Notary Public

"OFFICIAL SEAL"
TIMOTHY A. BYAM
NOTARY PUBLIC STATE OF ILLINOIS
COMMISSION EXPIRES 12/04/05

ATTACHMENT 2
Evaluation of Proposed Change

ATTACHMENT 2
Evaluation of Proposed Change

1.0 INTRODUCTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The proposed change revises Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A test to no later than July 22, 2009, for Unit 1, and no later than May 16, 2008, for Unit 2.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change involves a one-time exception to the 10 year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0 (Reference 1). Specifically, the proposed change revises TS 5.5.12 of the QCNPS, Units 1 and 2, TS to add the following statement:

", 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."

Attachment 3 provides a TS page markup indicating the proposed change. Attachment 4 provides the retyped TS pages incorporating the proposed change.

3.0 BACKGROUND

QCNPS, Units 1 and 2, are General Electric BWR/3 plants with Mark I primary containments. The Mark I primary containment consists of a drywell, which encloses the reactor vessel, reactor coolant recirculation system, and branch lines of the reactor coolant system (RCS); a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by access, piping, and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak tight characteristics of the primary containment at the design basis accident pressure.

Revisions to 10 CFR 50, Appendix J (i.e., Option B) allow individual plants to extend Type A ILRT surveillance testing frequency 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 is less than the maximum allowable primary containment leakage rate. In Reference 2, NRC approval to implement Option B was requested. The NRC subsequently approved implementation of Option B in Reference 3.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01. This document was established in 1995 during development of the performance-based Option B. Section 11.0 states that NUREG-1493, "Performance-Based Containment Leak Test Program," dated September 1995 (Reference 4), 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 Electrical Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals" (Reference 5).

Option B of 10 CFR 50, Appendix J, requires that a Type A test be conducted at a periodic interval based on historical performance of the overall primary containment system. QCNPS TS Section 5.5.12 provides the requirements for the Primary Containment Leakage Rate Testing Program. TS 5.5.12.a requires that this program 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. Additionally, the testing is required to conform to the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995 (Reference 6). Reference 6 endorses, with certain exceptions, NEI 94-01.

NEI 94-01 specifies for Type A tests, an initial test interval of 48 months and allows an extension of the interval to 10 years, based on two consecutive successful tests. QCNPS, Units 1 and 2, are currently on 10-year testing intervals.

The proposed change adds two exceptions to TS 5.5.12 to allow a one-time deferral from the guidelines contained in Regulatory Guide 1.163 and NEI 94-01 regarding the Type A test interval. The proposed change will extend the next Type A test for Units 1 and 2 to a 15-year interval.

The last Type A test for Unit 1 was performed on July 23, 1994, and the last Type A test for Unit 2 was performed on May 17, 1993. The proposed change will require the next Type A tests be performed by July 22, 2009, for Unit 1, and by May 16, 2008, for Unit 2.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in a licensee's TS.

10 CFR 50, Appendix J, Section V.B, "Implementation," specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant TS.

5.0 TECHNICAL ANALYSIS

5.1 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the primary containment, including systems and components that penetrate the primary containment, does not exceed allowable leakage rate values specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumptions in the safety analyses are not exceeded. The limitation of primary containment leakage provides assurance that the primary containment would perform its design function following an accident, up to and including the design basis accident.

10 CFR 50, Appendix J, was revised effective October 26, 1995. The purpose of this revision was to allow licensees to choose primary containment leakage testing under Option A "Prescriptive Requirements" or Option B. Amendment Nos. 169 and 165 for Units 1 and 2 (Reference 3) were issued to permit implementation of 10 CFR 50, Appendix J, Option B. TS 5.5.12 currently requires the establishment of a Primary Containment Leakage Rate Testing Program in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program implements the guidelines contained in Regulatory Guide 1.163, which specifies a method acceptable to the NRC for complying with Option B by approving the use of NEI 94-01, subject to several regulatory positions stated in the regulatory guide.

Deviations from Regulatory Guide 1.163 are permitted by 10 CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10 CFR 50, Appendix J, Option B.

Adoption of the Option B performance-based primary containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed or its acceptance criteria; however, it did alter the frequency at which Type A, B, and C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews as-found and as-left leakage history to determine the frequency for leakage testing, which provides assurance that leakage limits will be maintained.

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. NUREG-1493 made the following observations with regard to decreasing the test frequency.

- Reducing the Type A testing frequency to once per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is small because Type A tests identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have only been marginally above the existing requirements. Given the insensitivity of risk to primary containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between Type A testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (i.e., greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing requirements, the overall effect is very small.

NEI 94-01 requires that Type A testing be performed at lease once per 10 years based upon an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart or refueling cycles where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

5.2 QCNPS Integrated Leak Rate Testing History

Type A testing is performed to verify the integrity of the containment structure in its loss-of-coolant accident (LOCA) configuration. Industry test experience has demonstrated that Type B and C testing detects a large percentage of containment leakages and that the percentage of containment leakages detected only by integrated containment leakage testing is very small. Results of the previous five Type A tests for each unit, presented below, demonstrate the QCNPS Units 1 and 2 containment structures remain essentially leak tight barriers and represent minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493.

QCNPS Units 1 and 2 Type A Tests			
Unit	**Test Date**	**Total Leakage***	**Acceptance Limit***
1	March 22-23, 1986	0.2975	1.000
1	December 5-6, 1987	0.3508	1.000
1	November 14-15, 1989	0.4480	1.000
1	December 5-8, 1992	0.2944	1.000
1	July 23-24, 1994	0.3382	1.000
2	May 26-28, 1985	0.4092	1.000
2	October 14-15, 1986	0.3618	1.000
2	June 12-13, 1988	0.4621	1.000
2	April 27-28, 1990	0.4452	1.000
2	May 17-19, 1993	0.5064	1.000

* Leakage rates are expressed in units of percent containment air weight per day. The maximum allowable primary containment leakage rate, L_a, is 1% of primary containment air weight per day. TS leakage rate acceptance criteria for a Type A test for unit startup is $0.75L_a$ (i.e., 0.75% containment air weight per day), as discussed in TS 5.5.12. Calculated results are expressed at a 95% confidence level.

5.3 Type B and C Testing

Type B and C testing assures containment penetrations such as flanges, sealing mechanisms, and containment isolation valves are essentially leak tight. Type B and C tests identify the vast majority of all potential leakage paths. The Type B and C testing requirements will not be changed as a result of the extended ILRT interval.

5.4 Containment Inspections

a. Appendix J Visual Inspections

The Appendix J program requires visual inspections to be performed of accessible interior and exterior surfaces of the containment system for structural problems that may affect either the containment structural leakage integrity or performance of the Type A test. These examinations are conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test, based on a 10-year Type A test frequency (Reference 6).

These requirements will not be changed as a result of the extended ILRT interval.

b. Containment Inservice Inspection Program

A comprehensive primary containment inspection is performed to the requirements of American Society of Mechanical Engineers (ASME) Section XI, "Inservice Inspection," Subsections IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements of Class CC Concrete Components of Light-Water Cooled Power Plants."

The components subject to Subsection IWE and IWL requirements are those which make up the containment structure, its leak-tight barrier (including integral attachments), and those that contribute to its structural integrity. Specifically included are Class MC pressure retaining components, including metallic shell and penetration liners of Class CC pressure retaining components, and their integral attachments. QCNPS has no Class CC components which meet the criteria of IWL-1100, therefore, no requirements to perform examinations in accordance with Subsection IWL are incorporated into the QCNPS Containment Inservice Inspection (CISI) plan. The ASME Code Inspection Plan was developed in accordance with the requirements of the 1992 Edition with the 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWE and IWL, as modified by NRC final rulemaking to 10 CFR 50.55a published in the Federal Register on August 8, 1996.

The first interval of the QCNPS CISI Program is effective from September 30, 1998, through September 9, 2008. The first period of the first CISI interval ended on September 9, 2001, and the Unit 1 and Unit 2 CISI inspections for the first period have been completed. The second period of the first CISI interval is scheduled to end on September 9, 2005. For CISI inspections performed, various indications were observed, documented, evaluated, and determined to be acceptable. No areas of the containment liner surfaces require augmented examination, and no loss of structural integrity of primary containment was observed.

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

c. Containment Coatings Inspections

A program to maintain containment coatings was developed to meet the guidelines of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," Revision 0 (Reference 7). Each refueling outage, a preventive maintenance activity to inspect the protective coatings in the containment is performed. The most recent inspections for Units 1 and 2 were performed in November 2002, and February 2002, respectively. There have been minor issues noted (e.g., coating peeling); however, overall the containment coatings are in an acceptable condition.

The inspection requirements of the containment coatings program will not be changed as a result of the extended ILRT interval.

5.5 Information Notice 92-20

Information Notice 92-20, "Inadequate Local Leak Rate Testing," (Reference 8) was issued to alert licensees of problems with local leak rate testing of two-ply stainless steel bellows. The information notice discusses an event at QCNPS Unit 1 where a Type B test on the containment penetration bellows could identify leakage, but could not accurately quantify the extent of the leakage.

In Reference 9, the NRC granted an exemption from certain Type B testing requirements of 10 CFR 50, Appendix J, for the two-ply containment penetration expansion bellows at QCNPS. The exemption was needed because the bellows design is such that they cannot be properly tested to satisfy Type B testing requirements, barring replacement with bellows of a different design.

The exemption specifies an alternative program of bellows testing and replacement that involves testing with air at a reduced leakage limit, testing any leaking bellows with helium (i.e., sniffer testing), replacing bellows that are unacceptable, and performing a Type A test each refueling outage until all of the bellows have been replaced with testable bellows. This testing program is intended to assure that at least one ply of a two-ply bellows is intact and that overall containment leakage is within its allowable limit as shown by Type A testing. Reference 9 stated that the Type A test is essential to this program because it is the only test available that can properly quantify the bellows' leakages, albeit not individually. This is especially important for those bellows which are known to leak but will not be replaced until after another cycle.

As stated in Reference 9, the NRC found that the proposed testing program will detect bellows assemblies with significant flaws and result in replacement of flawed assemblies within one operating cycle, during which period there is reasonable assurance that the bellows assemblies will not suffer excessive degradation.

In Reference 10, Commonwealth Edison Company requested a revision to the exemption granted in Reference 9. The revised exemption would delete the requirement to perform a Type A test each refueling outage based on alternative Type B tests that were developed, since the original exemption was issued, to determine the leakage from the two-ply containment penetration expansion bellows. These alternative tests can be applied to ensure the intent of requiring a Type A test, as part of the original exemption, is met. As stated in Reference 10, the requirement to perform a Type A test every outage is not necessary to ensure that the bellows assemblies are adequately tested and leakage from any leaking bellows assembly is adequately quantified. This position was developed based upon the following insights gained during testing of two-ply bellows:

- there is minimal probability for the occurrence of a large leak in a two-ply bellows;
- the special testing program is effective for identifying small leaks in two-ply bellows;
- the Type A test is ineffective for identifying small leaks in two-ply bellows; and
- more cost effective alternative methods have been developed for quantifying leakage.

For a two-ply bellows that leaks through both plies, the revised exemption allows: (1) a valid Type B test using one of the alternative tests to ensure compliance to license limits, or (2) a Type A test as required in the original exemption and, before the return to power in a subsequent refuel outage, replacement of the bellows with a testable bellows assembly or a valid Type B test to ensure license limits are met.

In Reference 11, the NRC granted the revised exemption. As stated in Reference 11, the NRC found that the underlying purpose of the regulation will be met in that the proposed testing program will detect bellows assemblies with significant flaws and result in replacement of flawed assemblies within one operating cycle, or be tested with a Type B test to ensure license limits are met during which period there is reasonable assurance that the bellows assemblies will not suffer excessive degradation.

The proposed change to extend the Type A test frequency from once in 10 years to once in 15 years does not affect the conclusions documented in References 9 and 11. The NRC-approved testing program will continue to detect bellows assemblies with significant flaws and result in replacement of flawed assemblies within one operating cycle, or be tested with a Type B test to ensure license limits are met during which period there is reasonable assurance that the bellows assemblies will not suffer excessive degradation.

5.6 Risk Information

A plant-specific, risk-based evaluation has been performed in support of the one-time deferral of the Type A test frequency from once in 10 years to once in 15 years. The risk analysis is contained in Attachment 5. The 2002B QCNPS Unit 1 Level 1 and 2 probabilistic safety assessment (PSA) model was used as input to this analysis and is characteristic of the as-built, as-operated plant. There are no substantive differences between Unit 1 and Unit 2 that are judged to affect the conclusions of the PSA. As such, no separate PSA quantification was conducted for Unit 2. Since the PSA is judged applicable to both Units 1 and 2, the ILRT interval extension risk evaluation is applicable to both units

The risk analysis determined that the proposed change results in the following.

- Increasing the current 10 year ILRT interval to 15 years results in an insignificant increase in total population dose rate of 0.2%.
- The increase in the large early release frequency (LERF) risk measure is also insignificant, a 5.4E-9/year increase. This LERF increase is categorized as a "very small" increase per NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, November 2002 (Reference 12).
- Likewise, the conditional containment failure probability (CCFP) increases insignificantly by 0.3%.

Regulatory Guide 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Regulatory Guide 1.174 defines very small changes in risk as resulting in increases of core damage frequency below 10^{-6}/year and increases in LERF below 10^{-7}/year. Since the ILRT does not impact core damage frequency, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once per 10 years to once per 15 years, using the change in the EPRI Category 3b frequency per the NEI Interim Guidance (Reference 13), is 5.4E-9/year. Regulatory Guide 1.174 defines very small changes in LERF as below 10^{-7}/year. Therefore, increasing the QCNPS ILRT interval from 10 to 15 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

The change in CCFP is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The change in CCFP is found to be very small (i.e., 0.3% increase) and represents a negligible change in the QCNPS defense-in-depth.

The change in population dose rate is also reported consistent with previously approved ILRT interval extension requests. The change in population dose rate from the current once per 10 years to once per 15 years frequency is an insignificant 0.001 person-rem/year increase.

5.7 Conclusion

Based on the above, the proposed change to TS 5.5.12 will continue to provide assurance that leakage through the QCNPS primary containment will not exceed allowable leakage rate values specified in the TS and Bases, and that the containment features will continue to perform their design function following an accident, up to and including the design basis accident.

6.0 REGULATORY ANALYSIS

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

Additionally, 10 CFR 50, Appendix J, Section V.B, specifies that the regulatory guide or other implementing documents used to develop a performance-based leakage testing program must be included, by general reference, in the plant's TS. Additionally, deviations from guidelines endorsed in a regulatory guide are to be submitted as a revision to the plant's TS.

The proposed change will revise TS 5.5.12 to reflect a one-time deferral from the program requirements for the Type A test for QCNPS, Units 1 and 2. The deferral represents an exception to the guidelines contained in Regulatory Guide 1.163 and NEI 94-01. Thus, the proposed change is consistent with the requirements of 10 CFR 50.36(c)(5) and 10 CFR 50, Appendix J, Section V.B.

Additionally, in accordance with 10 CFR 50, Appendix J, Section V.B, the proposed changes to QCNPS TS do not require a supporting request for an exemption to Option B of Appendix J, in accordance with 10 CFR 50.12, "Specific exemptions."

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

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 an 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 to the TS for QCNPS, Units 1 and 2, 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 change will revise Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, Technical Specification (TS) 5.5.12, "Primary Containment Leakage Rate Testing Program," to reflect a one-time deferral of the primary containment Type A test to no later than July 22, 2009, for Unit 1, and no later than May 16, 2008, for Unit 2. The current Type A test interval of 10 years, based on past performance, would be extended on a one-time basis to 15 years from the last Type A test.

The function of the primary containment is to isolate and contain fission products released from the reactor coolant system (RCS) following a design basis loss of coolant accident (LOCA) and to confine the postulated release of radioactive material to within limits. The test interval associated with Type A testing is not a precursor of any accident previously evaluated. Therefore, extending this test interval on a one-time basis from 10 years to 15 years does not result in an increase in the probability of occurrence of an accident. The successful performance history of Type A testing provides assurance that the QCNPS primary containments will not exceed allowable leakage rate values specified in the TS and will continue to perform their design function following an accident. The risk assessment of the proposed change has concluded that there is an insignificant increase in total population dose rate and an insignificant increase in the conditional containment failure probability.

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 change for a one-time extension of the Type A tests for QCNPS, Units 1 and 2, will not affect the control parameters governing unit operation or the response of plant equipment to transient and accident conditions. The proposed change does not introduce any new equipment or modes of system operation. No installed equipment will be operated in a new or different manner. As such, no new failure mechanisms are introduced.

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

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

Response: No

QCNPS, Units 1 and 2, are General Electric BWR/3 plants with Mark I primary containments. The Mark I primary containment consists of a drywell, which encloses the reactor vessel, reactor coolant recirculation system, and branch lines of the RCS; a toroidal-shaped pressure suppression chamber containing a large volume of water; and a vent system connecting the drywell to the water space of the suppression chamber. The primary containment is penetrated by access, piping, and electrical penetrations.

The integrity of the primary containment penetrations and isolation valves is verified through Type B and Type C local leak rate tests (LLRTs) and the overall leak tight integrity of the primary containment is verified by a Type A integrated leak rate test (ILRT) as required by 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." These tests are performed to verify the essentially leak tight characteristics of the primary containment at the design basis accident pressure. The proposed change for a one-time extension of the Type A tests do not affect the method for Type A, B, or C testing, or the test acceptance criteria. In addition, based on previous Type A testing results, EGC does not expect additional degradation, during the extended period between Type A tests, which would result in a significant reduction in a margin of safety.

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

Based upon the above, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

8.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or would change an inspection or surveillance requirement. However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "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 need be prepared in connection with the proposed amendment.

9.0 PRECEDENT

The proposed amendment incorporates into the QCNPS TS changes that are similar to changes approved by the NRC for Susquehanna Steam Electric Station on March 8, 2002, Seabrook Station on April 11, 2002, and the Brunswick Steam Electric Plant, Units 1 and 2, on March 6, 2002, and November 21, 2002, respectively.

10.0 IMPACT ON PREVIOUS SUBMITTALS

EGC has reviewed the proposed change for impact on previous submittals awaiting NRC approval for QCNPS, and has determined that there is an impact to a submittal dated October 10, 2002 (Reference 14). The change proposed in Reference 14 revises TS page 5.5-12 to increase the maximum allowable primary containment leakage rate specified in TS 5.5.12.c from 1% of primary containment air weight per day to 3% of primary containment air weight per day. The retyped TS pages provided in Attachment 4 of this submittal **do not** reflect the change proposed in Reference 14.

EGC has performed a sensitivity study to evaluate the impact on the risk evaluation contained in Attachment 5 due to the proposed increased maximum allowable primary containment leakage rate. There is no impact on the large early release frequency or conditional containment failure probability. However, increasing the current 10 year ILRT interval to 15 years, assuming a maximum allowable primary containment leakage rate of 3% of primary containment air weight per day, results in an insignificant increase in total population dose rate of 0.4%.

11.0 REFERENCES

1. Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 0, July 26, 1995

2. Letter from P. L. Piet (Commonwealth Edison Company) to U. S. NRC, "Request for Amendment to Facility Operating Licenses NPF-11, NPF-18, DPR-19, DPR-25, DPR-29 and DPR-30, Appendix A, Technical Specifications, Incorporation of Option B to 10CFR50, Appendix J," November 14, 1995

3. Letter from J. F. Stang (U. S. NRC) to D. L. Farrar (Commonwealth Edison Company), "Issuance of Amendments Related to 10 CFR Part 50, Appendix J, Option B (TAC Nos. M94061, M94062, M94065, and M94066)," January 11, 1996

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

5. Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," August 1994

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

7. NRC Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," June 1973

8. NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing," March 3, 1992

9. Letter from B. A. Boger (U. S. 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)," February 6, 1992

10. Letter from J. L. Schrage (Commonwealth Edison Company) to W. T. Russell (U. S. NRC), "Request to Revise Exemption from 10CFR50 Appendix J Type B Testing Requirement for Two-Ply Containment Penetration Bellows," October 4, 1994

11. Letter from R. M. Pulsifer (U. S. NRC) to D. L. Farrar (Commonwealth Edison Company), "Revision to Exemption from Appendix J to 10 CFR Part 50 for Quad Cities, Units 1 and 2, and Dresden, Units 2 and 3 (TAC Nos. M90628, M90629, M90630 and M90631)," February 9, 1995

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

13. Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals," November 13, 2001

14. Letter from K. R. Jury (Exelon Generation Company) to U. S. NRC, "Request for License Amendments Related to Application of Alternative Source Term," October 10, 2002

ATTACHMENT 3
Markup of Proposed Technical Specifications Page Change

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

REVISED TECHNICAL SPECIFICATIONS PAGE

5.5-11

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.

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. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995. INSERT

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

(continued)

Insert for Technical Specification 5.5.12

, 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.

ATTACHMENT 4
Retyped Technical Specifications Pages for Proposed Change

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

REVISED TECHNICAL SPECIFICATIONS PAGES

5.5-11
5.5-12

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

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

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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. 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 1% 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.

ATTACHMENT 5

ERIN Report No. C46702044-5163, "Quad Cities Risk Assessment to Support ILRT (Type A) Interval Extension Request," December 2002

ERIN Engineering And Research, Inc

QUAD CITIES RISK ASSESSMENT TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

ERIN Report No. C46702044-5163

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DECEMBER 2002

QUAD CITIES RISK ASSESSMENT TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

ERIN Report No. C46702044-5163

Prepared by: S.A. Teagarden Date: 12/19/02

Reviewed by: [signature] Date: 12/26/02

Approved by: E.T. Burns Date: 12/28/02

Accepted by: [signature] Date: 1/2/03

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date
1	CITE Reference to LATEST RG 1.174	S.A. Teagarden 1/28/03	[signature] 1/28/03	E.T. Burns 1-29-03

TABLE OF CONTENTS

EXECUTIVE SUMMARY

The risk impact of a one-time extension of the Quad Cities (QC) integrated leak rate test (ILRT) interval from the currently approved once in 10 years to once in 15 years is evaluated. The results demonstrate that a change in the ILRT test interval from 10 years to 15 years represents a "very small" impact on risk, as defined by Reg. Guide 1.174.

The Quad Cities ILRT risk assessment uses Quad Cities specific information to calculate the existing risk profile and the changes to the risk profile for radionuclide releases. The ex-plant consequences are then calculated by adjusting the ex-plant consequences from a surrogate Mark I plant (as allowed by the NEI Interim Guidance). The evaluation utilizes NUREG/CR-4551 50-mile dose risk for a Mark I plant (Peach Bottom). The total dose risk is subdivided into accident progression bins (APBs) based on NUREG/CR-4551. The dose risk for each APB is adjusted to account for population differences, containment leakage rate, and power level for applicability to Quad Cities. The Quad Cities Level 2 release sequences are sorted to match the APBs and determine the Quad Cities specific accident frequency for each APB.

The Quad Cities accident frequency and dose for each APB is then converted to an equivalent EPRI category for consideration of the effects of ILRT interval changes. Three of the EPRI categories are affected by ILRT interval changes (1, 3a, and 3b). Table ES-1 summarizes the results.

The evaluation approach for the assessment of the risk is based on EPRI-TR-104285, NEI Interim Guidance (dated November 2001), and previous ILRT risk assessment submittals.

Three risk metrics are evaluated using the Quad Cities 2002B internal events PSA model:

Risk Metrics	Risk Increase
• Change in Large Early Release Frequency (LERF)	5.4E-9/yr
• Change in conditional containment failure probability	0.3%
• Change in population dose rate (person-rem/yr)	0.001

The first risk measure change is considered by Reg. Guide 1.174 as a "very small" impact on risk. The other two risk measure changes do not have criteria in Reg. Guide 1.174, but based on past ILRT interval extension requests these changes are also considered to represent "very small" impacts on risk.

Table ES-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Category Description	Dose (Person-Rem Within 50-miles)(1)	Quantitative Results as a Function of ILRT Interval			
			Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
			Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50-miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/ Year Within 50-miles)
1	No Containment Failure(2)	1.80E+3	3.78E-7	6 80E-4	3 18E-7	5.72E-4
2	Containment Isolation System Failure	5.88E+5	3 88E-9	2.28E-3	3 88E-9	2.28E-3
3a	Small Pre-Existing Failures(2), (3)	1 80E+4	1.09E-7	1.96E-3	1.63E-7	2.93E-3
3b	Large Pre-Existing Failures(2), (3)	6 30E+4	1.09E-8	6.84E-4	1 63E-8	1.03E-3
4	Type B Failures (LLRT)	N/A	N/A	N/A	N/A	N/A
5	Type C Failures (LLRT)	N/A	N/A	N/A	N/A	N/A
6	Other Containment Isolation System Failure	N/A	N/A	N/A	N/A	N/A
7	Containment Failure Due to Severe Accident	4 42E+5	1.58E-6	6.99E-1	1.58E-6	6.99E-1
8	Containment Bypass Accidents	5.88E+5	1.75E-8	1.03E-2	1.75E-8	1.03E-2
TOTALS:			2.10E-6(4)	7.15E-1	2.10E-6(4)	7.16E-1
Increase in Dose Rate						0 001
Increase in LERF					5 4E-9	
Increase in CCFP (%)					0.3%	

<u>Notes to Table ES-1:</u>

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant, Peach Bottom.

(2) Only EPRI categories 1, 3a, and 3b are affected by ILRT (Type A) interval changes.

(3) Dose estimates for categories 3a and 3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) Due to the NEI methodology and round off, the total frequency of all severe accidents is slightly less (approximately 4%) than the Quad Cities reported CDF.

Section 1
INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Quad Cities (QC) containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3], NEI Additional Information for ILRT Extensions [21], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PSA) findings and risk insights in support of a request for a change in a plant's licensing basis as outlined in Regulatory Guide 1.174 [4].

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 is less than normal containment leakage of 1.0La (allowable leakage).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5], "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], 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. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on Quad Cities specific models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November and December 2001, NEI issued enhanced guidance (hereafter referred to as the NEI Interim Guidance) that builds on the EPRI TR-104285 methodology and is intended to provide for more consistent submittals to the NRC. [3,21] The NEI Interim Guidance was developed for NEI by EPRI using personnel who also developed the EPRI TR-104285 methodology. This Quad Cities ILRT interval extension risk assessment employs the NEI Interim Guidance methodology.

It should be noted that, in addition to ILRT tests, containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency. Type C tests are also not affected by the Type A test frequency change.

1.3 CRITERIA

Based on previously approved ILRT extension requests, Quad Cities uses the following risk metrics to characterize the change in risk associated with the one time ILRT extension:

- Change in Large Early Release Frequency (LERF)
- Change in conditional containment failure probability
- Change in population dose rate (person-rem/yr)

Consistent with the NEI Interim Guidance, the acceptance guidelines in Regulatory Guide 1.174 [4] are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability, which helps to ensure that the defense-in-depth philosophy is maintained, will also be calculated.

In addition, based on the precedent of other ILRT extension requests [6], the total annual risk (person-rem/yr population dose rate) and the conditional containment failure probability are examined to demonstrate the relative change in risk. (No threshold has been established for these parameter changes.)

Section 2
METHODOLOGY

This section provides the following methodology related items:

- A brief summary of available resource documents to support the methodology
- The NEI Interim Guidance for the analysis approach to be used
- The assumptions used in the evaluation
- The inputs required
  - Generic ex-plant consequence
  - Plant specific inputs

The following subsections address these items.

2.1 GENERAL RESOURCES AVAILABLE

This section summarizes the general resources available as input. 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 [8]
6) NUREG-1493 [5]
7) EPRI TR-104285 [2]
8) NEI Interim Guidance [3, 21]
9) NUREG-1150 [14] and NUREG/CR-4551 [9]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA 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 and eighth studies are EPRI studies of the impact of extending ILRT and LLRT test intervals on at-power public risk. The ninth study provides consequence evaluations that can be used as surrogate results when corrected for QC specific characteristics.

<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 [15] 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 (PNL) 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. The study calculated unavailabilities for Technical Specification leakages and "large" leakages.

NUREG/CR-4220 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 [8]

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 is realized from extending the test intervals. For the BWR, the benefit from extending the ILRT frequency from 3 per 10 years to 1 per 10 years was calculated to be a reduction of approximately 1E-7/yr in the shutdown core damage frequency. This risk reduction is due to the following issues:

- Reduced opportunity for draindown events
- Reduced time spent in configurations with impaired mitigating systems

The study identified 7 shutdown incidents (out of 463 reviewed) that were caused by ILRT or LLRT activities. Two of the 7 incidents were RCS draindown events caused by ILRT/LLRT activities. The other 5 events involved loss of RHR and/or SDC due to ILRT/LLRT activities. This information was used in the EPRI study to estimate the safety benefit from reductions in testing frequencies. This represents a valuable insight into the <u>improvement</u> in the safety due to extending the ILRT test interval.

<u>NUREG-1493 [5]</u>

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

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population 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 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight (8) 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. For example, for the PWR analyzed, the change is about 0.02 person-rem per year . . ."

NEI Interim Guidance [3, 21]

NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions of Containment Integrated Leakage Rate Test Surveillance Intervals" [3] has been developed to provide utilities with revised guidance regarding licensing submittals. Additional information from NEI on the "Interim Guidance" was supplied in Reference [21].

A nine step process is defined which includes changes in the following areas of the previous EPRI guidance [2]:

- Impact of extending surveillance intervals on dose
- Method used to calculate the frequencies of leakages detectable only by ILRTs
- Provisions for using NUREG-1150 dose calculations to support the population dose determination.

This NEI Guidance is used in the Quad Cities ILRT analysis.

NUREG-1150 [14] and NUREG/CR 4551 [9]

NUREG-1150 and the technical basis, NUREG/CR-4551, provide an ex-plant consequence analysis for a spectrum of accidents including a severe accident with the containment remaining intact (i.e., Technical Specification leakage). The ex-plant consequences from NUREG-1150 used for the Quad Cities ILRT evaluation are taken from Peach Bottom (another Mark I plant).

2.2 NEI INTERIM GUIDANCE

The Quad Cities risk assessment analysis uses the approach outlined in the NEI Interim Guidance. [3,21] The nine steps of the methodology are:

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in ILRT test frequency.

2. Determine the containment leakage rates for EPRI categories 1, 3a and 3b.

3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.

4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (3) by the associated frequency calculated in Step (1).

5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

6. Determine the population dose rate for the new surveillance intervals of interest.

7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

8. Evaluate the risk impact in terms of LERF.

9. Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate 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 eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The ninth and final step of the interim methodology calculates the change in containment failure probability. The NRC has previously accepted similar calculations (Ref. [7], referred to as conditional containment failure probability, CCFP) as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such this last step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174.

2.3 GROUND RULES

The following ground rules are used in the analysis:

- The Quad Cities Unit 1 Level 1 and Level 2 internal events PSA model provides representative results for the analysis. Due to the similarity of Units 1 and 2, the results of this Unit 1 assessment apply to Unit 2 as well.

- It is appropriate to use the Quad Cities internal events PSA 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 rate) will not substantially differ if fire and seismic events were to be included in the calculations.

- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8] as augmented by NEI Interim Guidance. [3, 21]

- Radionuclide release categories are defined consistent with the EPRI TR-104285 methodology. [2]

- The ex-plant consequence in terms of population dose results for the containment failures modeled in the PSA can be characterized by information provided in NUREG/CR-4551 [9]. They are estimated by scaling the NUREG/CR-4551 population dose results by power level, population, and Tech Spec leak rate differences for Quad Cities compared to the NUREG/CR-4551 Mark I reference plant, Peach

Bottom. Use of dose results for the 50-mile radius around the plant as a figure of merit in the risk evaluation is consistent with NUREG-1150, past ILRT frequency extension submittals, and the NEI Interim Guidance.

- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 1 sequences is 1 L_a (L_a is the Technical Specification maximum allowable containment leakage rate).
- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 3a sequences is 10 L_a.
- Per the NEI Interim Guidance [3], the representative containment leakage for EPRI Category 3b sequences is 35 L_a.
- EPRI Category 3b is conservatively categorized as LERF based on the previously approved methodology [3].
- The impact on population doses from Interfacing System LOCAs 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 ISLOCA contribution to population dose is fixed, no changes on the conclusions regarding increases in population dose from this analysis will result from this assumption.
- The containment isolation valve test frequency is not altered. Therefore, the reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

2.4 PLANT SPECIFIC INPUTS

The inputs to the risk assessment include the following:

- Past Quad Cities ILRT results to demonstrate the adequacy of the administrative and hardware issues.
- Ex-plant consequence evaluation from NUREG-1150 for a Mark I plant
- Quad Cities Unit 1 PSA Model 2002B (Level 1 & 2)

- Quad Cities specific adjustments to ex-plant consequence evaluation from NUREG-1150 (NUREG/CR 4551 Vol. 4 for Peach Bottom)

2.4.1 Prior Quad Cities ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

Based on the consecutive successful ILRTs performed in the early 1990's, the current ILRT interval for Quad Cities is once per ten years. [16]

2.4.2 Ex-Plant Consequences

Consistent with the NEI Interim Guidance [3] and the supplemental information [21], ex-plant consequence evaluations from NUREG-1150 can be used in the ILRT evaluation to support the population dose estimate.

Figure 2-1 is a simplified flow chart that shows the process for determining the Quad Cities specific population dose (person-rem) for comparable radionuclide release categories starting with the NUREG-1150 Mark I (Peach Bottom) ex-plant consequence evaluation and correcting for key differences.

The surrogate plant consequence analysis for Peach Bottom is calculated for the 50-mile radial area surrounding Peach Bottom (A). The ex-plant calculation is delineated by total person-rem for each identified Accident Progression Bin (APB) from NUREG/CR-4551 (B). The Quad Cities Level 2 model end states are assigned to one of the NUREG/CR-4551 APBs (C, E).

In order to convert the Peach Bottom population dose estimates for use in the Quad Cities consequence evaluation, adjustments to these ex-plant consequences are needed to account for the following (D, F, G):

- Population differences
- Containment leakage rate
- Power level

Finally, the Quad Cities specific ex-plant consequences are calculated (H).

The parameters that were used in the Peach Bottom analysis from NUREG/CR-4551 for comparison with Quad Cities are the following:

- Peach Bottom Population out to 50 miles = 3.2E+6 persons
  (See Appendix A for derivation)

- Peach Bottom Power level = 3293 MWt
  (See [9] Table 4.2-1)

- Peach Bottom Containment leak rate = 0.5%/day[(1)]

While meteorology could play a role in the early health effects calculations, the meteorology and site topography for Peach Bottom and Quad Cities are assumed to be sufficiently similar that these differences are assumed not to play a significant role in this evaluation of total population dose.

2.4.3 Plant Specific Inputs

The Quad Cities specific information used to perform this ILRT interval extension risk assessment includes the following:

- Quad Cities Unit 1 Level 1 PSA Model 2002B

(1) The analysis performed in NUREG/CR-4551 used a leakage of 0.5%/day (Vol 4, Rev. 1, Part 2, Page B.2-9). The current Peach Bottom Technical Specification leakage may differ.

- Quad Cities Unit 1 Level 2 PSA Model 2002B
- Quad Cities Plant and Site Characteristics
  - Population out to 50 miles = 7.0E+5 persons (year 2000)
    (See Appendix A for derivation)
  - Power Level = 2957 MWt
    (Includes recent 17% power uprate)
  - Containment Leakage Rate = 1.0% vol/day
    (Quad Cities Improved Technical Specification Bases, B 3.6.1.1)

2.4.3.1 Quad Cities Unit 1 PSA

The Quad Cities Unit 1 Level 1 and 2 PSA (2002B) used as input to this analysis is characteristic of the as-built, as-operated plant. The 2002B PSA model is the latest Quad Cities model with detailed Level 2 sequences. A 2002C model has been recently completed which incorporates Large Early Release Frequency (LERF) multipliers for the Level 2 analysis rather than detailed Level 2 event tree sequences. The 2002C Level 2 model is based on the 2002B Level 1 model CDF and is used to support on-line maintenance risk assessment. The 2002C model is not as conducive to the ILRT calculational methodology for non-LERF releases as the detailed Level 2 event tree sequences of the 2002B model. Since the Level 1 CDF is the same for both models, the 2002B model is acceptable for use.(1) The 2002B model is developed in CAFTA.

The QC total core damage frequency (CDF) as reported in the Quad Cities Level 2 Notebook is 2.18E-6/yr at a truncation of 5E-12/yr [18]. Table 2-1a summarizes the Quad Cities Level 1 PSA frequency results by core damage accident class. Table 2-1b summarizes the Quad Cities Level 2 PSA results for containment failure.

(1) The Quad Cities PSA 2002B model does not include an internal flood analysis. Internal flooding is included in the draft 2002D model. Appendix B addresses the impact of internal flooding on the ILRT results as part of the external event impact assessment.

The Quad Cities Level 2 PSA is used to calculate the release frequencies for the accidents evaluated in this assessment. The Level 2 PSA is also developed in CAFTA. Table 2-2 summarizes the pertinent Quad Cities Level 2 PSA results in terms of release categories. [18]. The total release frequency is 1.68E-6/yr; with a total CDF of 2.18E-6/yr. The "No Release" frequency (i.e., containment leakage within Technical Specifications) for Quad Cities is 4.97E-7/yr. [18]

2.4.3.2 Quad Cities Unit 2

No substantive differences exist between the Quad Cities Unit 1 and Unit 2 that are judged to affect the conclusions of the PSA. As such, no separate PSA quantification is conducted for Unit 2. Since the Quad Cities PSA is judged applicable to both Unit 1 and Unit 2, the ILRT internal extension evaluation based upon the Quad Cities PSA is considered applicable to both Unit 1 and Unit 2.

2.4.4. Adjustments to Ex-plant Consequence Calculations

This NUREG/CR-4551 ex-plant consequence analysis is calculated for the 50-mile radial area surrounding Peach Bottom, and is reported in total person-rem for discrete accident categories (termed Accident Progression Bins (APB) in NUREG/CR-4551). To use the NUREG/CR-4551 consequences in this ILRT risk assessment, the following steps should first be performed:

- Adjust the person-rem results to account for differences between the Peach Bottom analyses in NUREG/CR-4551 and the Quad Cities plant and its demographics:
  - Population
  - Reactor Power Level
  - Technical Specification Allowed Containment Leakage Rate
- Assign the adjusted NUREG/CR-4551 APB consequences to the EPRI categories used in this risk assessment

2.4.4.1 Surrounding Population

The 50-mile radius population used in the Peach Bottom NUREG/CR-4551 consequence calculations is 3.2E+6 persons (refer to Appendix A of this report).

For the Quad Cities population estimate, data is available for population by county from the US Census Bureau on the web site (http://eire.census.gov/popest/data/counties/tables). This data is used to estimate the population within a 50-mile radius of the plant. If the entire county falls within the 50-mile radius based on a review of a map containing a mileage scale and county borders, then the entire population is included in the population estimate. Otherwise, a fraction of the population is counted based on the percentage of

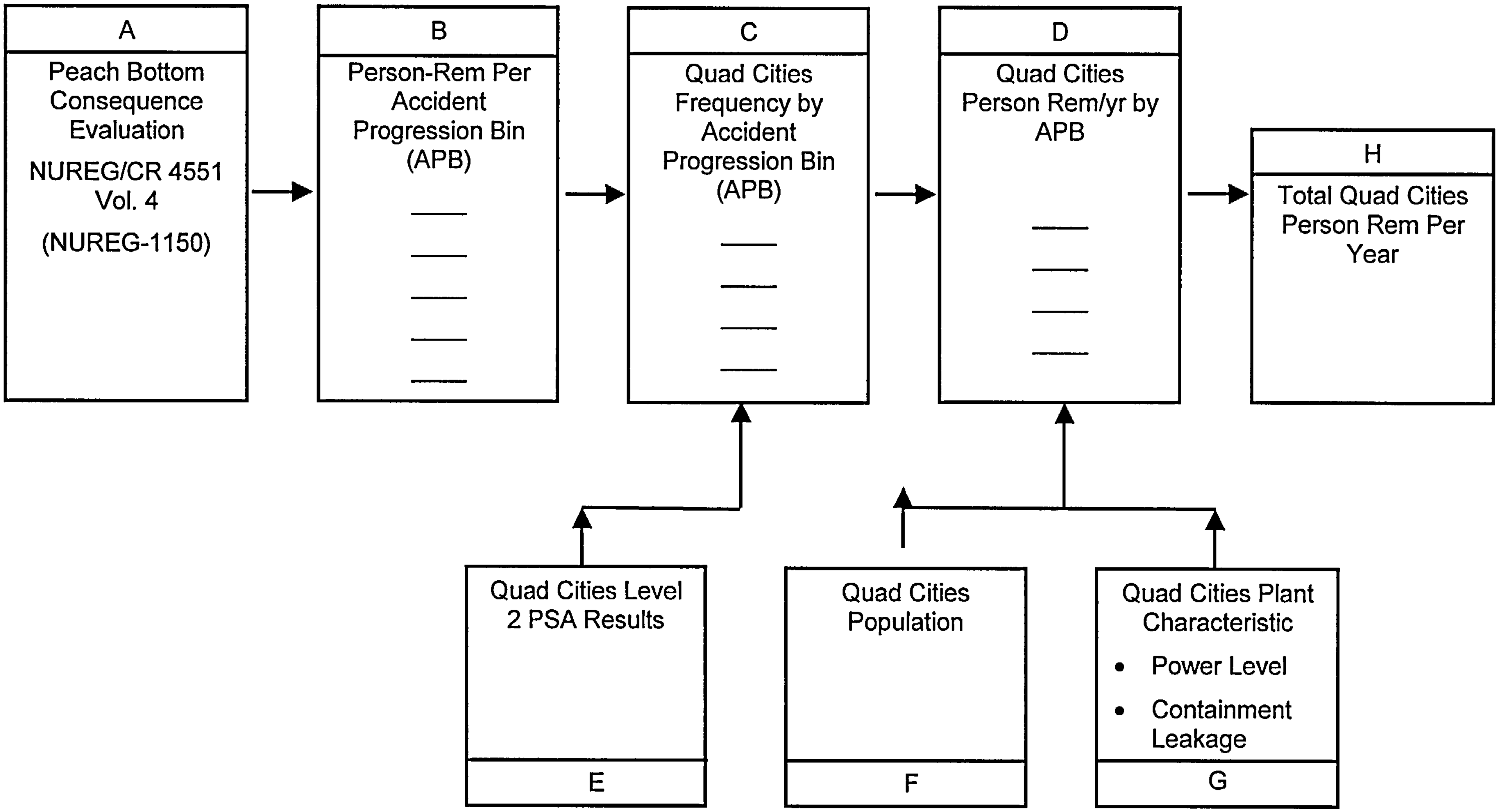

Figure 2-1 Process for Calculating Population Dose for Quad Cities Using the Surrogate Plant Results from NUREG-1150

the county within the 50-mile radius. The land area within the 50-mile radius is estimated based on visual inspection of the map and the population of that area is estimated assuming uniform distribution of the population within the county. The results of the population estimate for Quad Cities are presented in Table 2-3.

The year 2000 population within the 50-mile radius of Quad Cities is estimated in Appendix A of this report at 7.0E+5 persons.

The ratio of the population surrounding Quad Cities to that in the Peach Bottom analysis results in a factor of:

$$\frac{7.0\text{E+5persons}}{3.2\text{E+6 persons}} = 0.22$$

2.4.4.2 Reactor Power Level

The Peach Bottom reactor power level used in the NUREG/CR-4551 consequence calculations is 3293 MWt. Quad Cities recently performed a power uprate of 17% over the originally licensed thermal power; the projected Quad Cities full power level is 2957 MWt.

The Quad Cities power level used in this ILRT evaluation is the extended power uprate power level of 2957 MWt. This represents a factor of 0.90 = (2957 MWt/3293 MWt) change in the population dose for each APB.

2.4.4.3 Technical Specification Containment Leakage

The Peach Bottom analysis in NUREG/CR-4551 assumes containment leakage of 0.5% vol./day (see Vol. 4, Rev. 1, Part 2, page B.2-9).

The Quad Cities Technical Specification leakage is 1.0% vol./day. Because the leakage rates are in terms of the containment volume, these plant characteristics are also needed:

- Peach Bottom Containment Volume:

	min (ft^3)	max (ft^3)	average (ft^3)
Drywell free volume(1)	1.59E+5	1.76E+5	1.67E+5
Supp. Pool free volume(1)	1.28E+5	1.32E+5	1.30E+5
TOTAL	--	--	2.97E+5

- Quad Cities Containment Volume:

	min (ft^3)	max (ft^3)	average (ft^3)
Drywell free volume(2)	--	--	1.58E+5
Supp. Pool free volume(2)	1.11E+5	1.19E+5	1.15E+5
TOTAL	--	--	2.73E+5

(1) NUREG/CR-4551, Vol 4, Part 2, A 31.
(2) Quad Cities UFSAR, Table 6.2-1.

For this comparison, the following factor can be developed to relate the leakage[1] impact between the two plants:

$$\frac{\text{Total Leakage Quad Cities}}{\text{Total Leakage Peach Bottom}} = \frac{1.0 \text{ Vol.}^{QC} \text{ \%/day}}{0.50 \text{ Vol.}^{PB} \text{ \%/day}}$$

$$= \frac{1.0\% \text{ /day}}{0.50\% \text{ /day}} \Sigma \frac{1}{Z}$$

$$\frac{\text{Total Leakage for Quad Cities}}{\text{Total Leakage for Peach Bottom}} = 2.0 \Sigma 0.92 = 1.84$$

This represents a factor of 1.84 increase in the person-rem consequence for the "intact" containment APB.

2.4.4.4 Summary

The factors that are calculated for use in adjusting the population dose (person-rem) of the surrogate plant (NUREG-1150 Peach Bottom) for the site and plant differences are as follows:

- Consequence categories dependent on the "intact" Tech Spec Leakage

  $F_{CAT\ 1,\ 3a,\ 3b} = F_{POPULATION} \Sigma F_{POWER} \Sigma F_{TS\ LEAK}$

  $F_{CAT\ 1,\ 3a,\ 3b} = 0.22 \Sigma 0.90 \Sigma 1.84$

  $F_{CAT\ 1,\ 3a,\ 3b} = 0.36$

---

[1] Ratio of containment volumes is needed to relate the leakage rates:

$$\frac{Vol^{QC}}{Vol^{PB}} = \frac{1}{Z}$$

Where

$Vol^{PB} = Z\ Vol^{QC}$

$$Z = \frac{Vol^{PB}}{Vol^{QC}} = \frac{2.97E+5 \text{ ft}^3}{2.73E+5 \text{ ft}^3} = 1.09, \quad 1/Z = 0.92$$

Containment Vol. of Peach Bottom = 2.97E+5 ft^3

Containment Vol. of Quad Cities = 2.73E+5 ft^3

- Consequence categories not dependent on the Tech Spec Leakage:

  $F_C = F_{POPULATION}\ \Sigma\ F_{POWER}$

  $F_C = 0.22\ \Sigma\ 0.90$

  $F_C = 0.20$

Table 2-1a

CORE DAMAGE FREQUENCY CONTRIBUTIONS BY ACCIDENT CLASS [18][(2)]

Contributing Accident Class		Core Damage Frequency[(1)]	% of CDF
<u>Transients</u>			
Class IA/IE	Transients – Core Melt with Vessel at High Pressure	8.73E-7	40.1%
Class IC	ATWS with Loss of Injection	4.04E-9	0.2%
Class ID	Transients – Core Melt with Vessel at Low Pressure	1.21E-9	0.1%
Class II	Core Melts After Containment Failure Because of Loss of DHR Capability	6.45E-7	29.6%
<u>SBO</u>			
Class IBE	Station Black Out - Early	2.15E-8	1.0%
Class IBL	Station Black Out - Late	3 14E-7	14.4%
<u>LOCAs</u>			
Class 3B	Small or Medium LOCA – Core Melt with Vessel at High Pressure	1.34E-8	0.6%
Class 3C	Medium or Large LOCA – Core Melt with Vessel at Low Pressure	1.12E-7	5.1%
Class 3D	LOCA - Core Melt and Containment Failure Near Simultaneous from Vapor Suppression Failure	1.19E-8	0.5%
Class V	Interfacing System LOCA	1.75E-8	0.8%
<u>ATWS</u>			
Class IV	ATWS – Containment Fails Before Core Damage	1.64E-7	7.6%
Total		2.18E-6[(3)]	100%

(1) All frequencies in events per reactor year.

(2) Source: Table 7.2-2 of Volume 1.

(3) Total CDF is based on QC PSA Level 2 Notebook for model 2002B. [18] The QC PSA model does not include internal flooding The impact of internal flooding is addressed in Appendix B of this report.

Table 2-1b

SUMMARY OF QUAD CITIES LEVEL 2 PSA RESULTS [18][(2)]

End State	Core Damage Frequency (per year)[(1)]	Percent
Containment Intact (Tech Spec leakage)	4.97E-7	23%
Containment Failure (All other release categories)	1.68E-6	77%
Total	2.18E-6[(3)]	100%

(1) All frequencies in events per reactor year.

(2) Source: Table 7.2-2 of Volume 1

(3) Total CDF is based on QC PSA Level 2 Notebook for model 2002B [18] The QC PSA model does not include internal flooding. The impact of internal flooding is addressed in Appendix B of this report.

Table 2-2

SUMMARY OF QUAD CITIES PSA LEVEL 2 RESULTS [18][2]

Release Category	Frequency[1] (per year)
H/E -- High Early (LERF)	2 68E-7
M/E - Medium Early	2.49E-7
L/E - Low Early	9.72E-9
LL/E - Low Low Early	0
H/I - High Intermediate	3.55E-8
M/I - Medium Intermediate	7.96E-7
L/I - Low Intermediate	3.01E-7
LL/I - Low Low Intermediate	4.88E-9
H/L - High Late	5.47E-9
M/L - Medium Late	7.01E-11
L/L - Low Late	9 45E-9
LL/L - Low Low Late	1.54E-9
Total Release Frequency	**1.68E-6**
Core Damage Frequency	**2.18E-6[3]**

(1) All frequencies in events per reactor year.

(2) Source: Table 7.2-2 of Volume 1.

(3) Total CDF is based on QC PSA Level 2 Notebook for model 2002B. [18] The QC PSA model does not include internal flooding The impact of internal flooding is addressed in Appendix B of this report

Table 2-3

POPULATION WITHIN 50-MILES OF QUAD CITIES
(2000 US CENSUS)

County	2000 Census Population by County[4]	Percent Area of County in 50 Mile Radius[1]	Population within 50 Mile Radius[2]
Cedar, IA	18,187	80	14,550
Clinton, IA	50,149	100	50,149
Dubuque, IA	89,143	10	8,914
Jackson, IA	20,296	100	20,296
Jones, IA	20,221	40	8,088
Muscatine, IA	41,722	60	25,033
Scott, IA	158,668	100	158,668
Bureau, IL	35,503	60	21,302
Carroll, IL	16,674	100	16,674
Henry, IL	51,020	100	51,020
Jo Davies, IL	22,289	75	16,717
Knox, IL	55,836	35	19,543
Lee[3], IL	36,062	35	22,700[3]
Mercer, IL	16,957	90	15,261
Ogle, IL	51,032	35	17,861
Rock Island, IL	149,374	100	149,374
Stark, IL	6,332	50	3,166
Stephenson, IL	48,979	30	14,694
Warren, IL	18,735	10	1,874
Whiteside, IL	60,653	100	60,653
TOTALS:	967,832	--	696,557

(1) Based on visual inspection of Iowa and Illinois state maps.
(2) County Population multiplied by percentage within 50-mile zone, except when noted.
(3) Population density varied greatly in this region, an exception was made
(4) Source: http //eire census gov/popest/data/counties/tables

Section 3
ANALYSIS

This section provides a step-by-step summary of the NEI guidance as applied to the Quad Cities ILRT interval extension risk assessment. Each subsection addresses a step or group of steps in the NEI guideline.

3.1 BASELINE ACCIDENT CATEGORY FREQUENCIES (STEP 1)

The first step of the NEI Interim Guidance is to quantify the baseline frequencies for each of the EPRI TR-104285 accident categories. This portion of the analysis is performed using the Quad Cities Level 1 and Level 2 PSA results. The results for each EPRI category are described below.

Tables 2-1a, 2-1b and 2-2 of Section 2 compiled from the Quad Cities PSA [18] are used for the inputs to the accident frequency assessment.

Frequency of EPRI Category 1

This group consists of all core damage accident sequences in which the containment is initially isolated and remains intact throughout the accident (i.e., containment leakage at or below maximum allowable Technical Specification leakage). The ILRT methodology artificially divides this category among the Tech Spec leakage case (Category 1) and two other categories that are used to simulate possible changes due to reduced ILRT frequencies (i.e., Categories 3a and 3b; see below for their definition). Per NEI Interim Guidance, the frequency per year for this category is calculated by subtracting the frequencies of EPRI Categories 3a and 3b (see below) from the sum of all severe accident sequence frequencies in which the containment is initially isolated and remains intact (i.e., accidents classified as “OK” in the Quad Cities Level 2 PSA).

As discussed previously in Section 2.4, the frequency of the Quad Cities Level 2 PSA "OK" or "No Release" accident bin is 4.97E-7/yr. As described below, the frequencies of EPRI Categories 3a and 3b are 3.26E-8/yr and 3.26E-9/yr, respectively. Therefore, the frequency of EPRI Category 1 is calculated as 4.97E-7/yr – 3.26E-8/yr - 3.26E-9/yr = 4.61E-7/yr.

Frequency of EPRI Category 2

This group consists of all core damage accident sequences in which the containment isolation system function fails during the accident progression (e.g., due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause valve failures).

The frequency of this EPRI category is estimated by multiplying the conditional probability of containment isolation failure from the Quad Cities Level 2 PSA by the portion of the severe accident sequences (CDF) that would be challenged. The sequences that have containment isolation already failed are Class II, Class IIID, Class IV, and Class V. Therefore, the EPRI Category 2 CDF does not include QC Level 1 Class II, Class IIID, Class IV, or Class V accident sequences. The following values are used for this calculation:

- Containment Isolation System failure probability = 2.90E-3 [18] (1)
- Total CDF = 2.18E-6/yr [18]
- Class II sequences = 6.45E-7/yr [18]
- Class IIID sequences = 1.19E-8/yr [18]
- Class IV sequences = 1.70E-7/yr [18]
- Class V sequences = 1.75E-8/yr [18]

The frequency per year for this category is calculated as follows:

(1) Containment isolation system failure probability based on nodal quantification of event node IS2 for loss of offsite AC or DC Division I and II (7.90E-3) minus the pre-existing containment failure probability basic event (5.00E-3). Pre-existing containment failures are evaluated in other EPRI categories

Frequency 2 = (containment isolation failure probability)
x (CDF – CDF of Class II - CDF of Class IIID - CDF of Class IV - CDF of Class V)

Frequency 2 = (2.90E-3) X (2.18E-6/yr – 6.45E-7/yr - 1.19E-8/yr - 1.70E-7/yr - 1.75E-8/yr)

Frequency 2 = 3.88E-9/yr

Note that pre-existing isolation failures are included in Category 6.

The frequency of EPRI Category 2 is 3.88E-9/yr.

Frequency of EPRI Category 3a

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "small" leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Consistent with NEI Interim Guidance [21], the frequency per year for this category is calculated as:

Frequency 3a = (3a conditional failure probability) x (CDF - CDF with independent LERF - CDF that cannot cause LERF)

The 3a conditional failure probability (2.7E-2) value is the conditional probability of having a pre-existing "small" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

The pre-existing leakage probability is multiplied by the residual core damage frequency (CDF) determined as the total CDF minus the CDF for those individual sequences that either may already (independently) cause a LERF or could never cause a LERF due to the delay time of the release (i.e., non-early). As discussed previously in Section 2.4.2, the Quad Cities total core damage frequency is 2.18E-6/yr. Of this total CDF, the following core damage accidents involve either LERF directly (containment bypass) or will never result in LERF:

- Long Term Station Blackout (SBO) scenarios (Class IBL) = 3.14E-7/yr [18]
- Loss of Containment Heat Removal accidents (Class II): 6.45E-7/yr [18][(1)]
- Containment Bypass accidents (Class V): 1.75E-8/yr [18]

Therefore, the frequency of EPRI Category 3a is calculated as (2.70E-02) x (2.18E-6/yr – 3.14E-7/yr - 6.45E-7/yr - 1.75E-8/yr) = 3.26E-8/yr.

<u>Frequency of EPRI Category 3b</u>

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "large" leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Similar to Category 3a, the frequency per year for this category is calculated as:

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

The 3b failure probability (2.7E-3) value is the conditional probability of having a pre-existing "large" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

Therefore, similar to EPRI Category 3a, the frequency of Category 3b is calculated as (2.70E-03) x (2.18E-6/yr – 3.14E-7/yr - 6.45E-7/yr - 1.75E-8/yr) = 3.26E-9/yr.

Frequency of EPRI Category 4

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type B component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 5

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type C component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type C tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 6

This group consists of all core damage accident sequences in which the containment isolation function is failed due to "other" pre-existing failure modes (e.g., pathways left open or valves that did not properly seal following test or maintenance activities) that would not be identifiable by containment leak rate tests. Per NEI Interim Guidance, because this category of failures is not impacted by leak rate tests, this group is not evaluated further in this analysis.

Frequency of EPRI Category 7

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). Other severe

(1) Per Table 6.6.2 of [18], a small percentage (0.0326%) of Class II releases do contribute to LERF. This small contribution (6.45E-7/yr * 3.26E-4 = 2.10E-10/yr, 6 45E-7/yr - 2.10E-10/yr ≈ 6.45E-7/yr) however, is negligible and is omitted in the calculation.

accidents such as intact containment leakage and containment bypass are accounted for in other EPRI categories. Per NEI Interim Guidance, the frequency per year for this category is based on the plant Level 2 PSA results.

For this analysis, the associated radionuclide releases are based on the application of the Quad Cities Level 2 sequence end states to the Accident Progression Bins from NUREG/CR-4551. The collapsed APBs are characterized by five attributes related to the accident progression. These five characteristics are: core damage, vessel breach, containment failure timing, containment failure location (wetwell vs. drywell), and RPV pressure (high vs. low). The Quad Cities Level 2 PSA containment event tree sequences can be correlated or binned into similar groups that are then characterized in terms of release magnitude and ex-plant consequence as categorized in NUREG/CR-4551 for the surrogate Mark I plant. This binning matches the similarity in release path and scenario definition between the Quad Cities Level 2 PSA and NUREG/CR-4551. As such, EPRI Category 7 is divided into eight sub-categories which are directly mapped to Bins 1-7 and 9 from NUREG/CR-4551. (No Quad Cities sequence end states matched the definition for Bin 8).

Table 3-1 provides the NUREG/CR-4551 APB definitions. Table 3-2 summarizes the assumptions used to assign the Quad Cities Level 2 end states to the various APBs of NUREG/CR-4551. Table 3-3 summarizes the Quad Cities Level 2 sequences assigned to each APB.

The frequency of each Category 7 subgroup is as follows:

Category 7a	2.69E-9/yr
Category 7b	1.85E-9/yr
Category 7c	1.05E-7/yr
Category 7d	4.90E-7/yr
Category 7e	1.33E-9/yr
Category 7f	8.15E-7/yr
Category 7g	1.24E-7/yr
Category 7h	4.07E-8/yr
Total	1.58E-6/yr

Frequency of EPRI Category 8

This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., Break Outside Containment LOCA or Interfacing Systems LOCA, ISLOCA). The frequency of Category 8 is the total frequency of the Quad Cities Level 1 PSA containment bypass scenarios (Class V). Based on the Quad Cities Level 1 PSA results summarized earlier in Table 2-1, the frequency of Category 8 is 1.75E-8/yr.

Summary of Frequencies of EPRI Categories

In summary, per the NEI Interim Guidance, the accident sequence frequencies that can lead to radionuclide releases to the public have been derived for accident categories defined in EPRI TR-104285. The accident sequence frequency results by EPRI category are summarized in Table 3-4.

Table 3-1

COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [9]

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.

Table 3-1

COLLAPSED ACCIDENT PROGRESSION BIN (APB) DESCRIPTIONS [9]

Collapsed APB Number	Description
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics. The risk associated with this bin is negligible.

<u>Legend</u>

CD = Core Damage

VB = Vessel Breach

CF = Containment Failure

WW = Wetwell

DW = Drywell

RPV = Reactor Pressure Vessel

Table 3-2

QUAD CITIES LEVEL 2 MODEL NODAL ASSUMPTIONS FOR APPLICATION TO THE NUREG/CR-4551 ACCIDENT PROGRESSION BINS

QUAD CITIES PSA Containment Event Tree Node	Assumption
IS - Containment Isolation	If the containment is not isolated, it is assumed that it will be open for the equivalent of an un-scrubbed release as soon as the vessel is breached. No depressurization is asked prior to this node; it is assumed that RPV pressure is > = 200 psi for these sequences This is APB #3.
OP - Operator depressurizes the RPV	It is assumed that success on this branch results in RPV pressure below 200 psi that is then used to distinguish between APB #1 versus APB #2, or APB #3 versus APB #4.
RX - Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are grouped in APB #9. However, this assignment is overridden if the containment still fails due to subsequent CZ, SP, HR, VC or MU failures. In these cases, CZ failures are assigned to APB #3 or APB #4 depending on the status of OP, and APB #5 or APB #6 is assigned for SP, HR, VC or MU failures depending on the status of the SP and WW nodes.
CZ, DI, NC, SI - Containment Intact Nodes	Failure of containment is assumed to result in an un-scrubbed release The timing is assumed to be "early" for all but loss of containment heat removal (Level 2 Accident Class II) events and is grouped in APB #3 or APB #4 depending on RPV pressure. For the Level 2 Accident Class II events, the timing is assumed to be "late" and is grouped in APB #5 or #6 depending on whether the suppression pool is not bypassed in the DI, WW, and SI nodes.
FC - Containment Flooding	If containment flooding is initiated and successfully completed without other containment failures, this is assigned to APB #7 basd on the interpretation that the successful completion of flooding required RPV venting RPV venting is assumed to result in a release characteristic similar to the venting scenarios from APB #7.
CV, GV - Containment Venting Nodes	Success of these nodes is used to indicate assignment to APB #7 for venting as long as the suppression pool is not bypassed and other containment failure nodes are not failed. This assignment applies to sequences with RX failures
SP, WW - Suppression Pool Not Bypassed Nodes	The suppression pool bypass nodes are considered in the CETs to determine whether the vent volume passes through the suppression pool or not. This node is used to distinguish between a WW or DW failure as described in the other node assumption descriptions above.
RB - Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release (with the success of the SP node) and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub.

Table 3-3

QUAD CITIES LEVEL 2 CET SEQUENCE ASSIGNMENT TO APB

Accident Progression Bin	QC Level 2 Sequence(1)		Bin Total (per year)
	Sequence	Frequency (Per year)	
APB #1 (Category 7a)	IA2-26	2.30E-10	2.69E-09
	IA2-28	1.88E-09	
	IA2-35	5.76E-10	
APB #2 (Category 7b)	IA1-26	5.29E-11	1.85E-09
	IA1-27	4.45E-10	
	IA1-29	1.18E-10	
	IA1-31	9.60E-10	
	IA1-39	2.74E-10	
APB #3 (Category 7c)	3B-41	8.37E-12	1.05E-07
	3C-41	6.02E-10	
	IA1-43	6.18E-09	
	IA2-24	1.13E-10	
	IA2-25	8.67E-10	
	IA2-36	7.51E-08	
	IA2-37	1.45E-09	
	IBE1-43	1.04E-10	
	IBE2-36	4.14E-10	
	IBL1-43	2.07E-09	
	IBL2-18	6.80E-12	
	IBL2-36	1.29E-08	
	IBL2-37	1.72E-10	
	IVA1-28	1.64E-11	
	IVA2-24	2.28E-09	
	IVA2-50	2.28E-09	

Table 3-3

QUAD CITIES LEVEL 2 CET SEQUENCE ASSIGNMENT TO APB

Accident Progression Bin	QC Level 2 Sequence[1]		Bin Total (per year)
	Sequence	Frequency (Per year)	
APB #4 (Category 7d)	3B-20	8.29E-12	4.90E-07
	3B-38	3.89E-10	
	3C-38	1.09E-07	
	3C-39	9.85E-10	
	3D-09	1.19E-08	
	IA1-20	1.84E-09	
	IA1-40	3.42E-08	
	IA1-41	3.22E-09	
	IBE1-20	1.70E-11	
	IBE1-40	6.17E-09	
	IBE1-41	9.27E-11	
	IBL1-20	1.69E-10	
	IBL1-40	1.61E-07	
	IBL1-41	2.02E-09	
	ID1-40	1.92E-10	
	IVA1-14	1.37E-09	
	IVA2-11	7.80E-08	
	IVA2-12	4.68E-10	
	IVA2-36	7.80E-08	
	IVA2-37	4.68E-10	
APB #5 (Category 7e)	3B-06	9.81E-12	1.33E-09
	IA1-06	2.18E-11	
	IA1-07	1.81E-10	
	IA1-15	1.19E-10	
	II2-12	9.97E-10	

Table 3-3

QUAD CITIES LEVEL 2 CET SEQUENCE ASSIGNMENT TO APB

Accident Progression Bin	QC Level 2 Sequence[1] Sequence	QC Level 2 Sequence[1] Frequency (Per year)	Bin Total (per year)
APB #6 (Category 7f)	3B-05	1.21E-09	8.15E-07
	3B-09	4.08E-11	
	3B-11	2.68E-11	
	3B-27	3.29E-11	
	3B-29	2.93E-11	
	IA1-05	2.40E-07	
	IA1-11	8.25E-09	
	IA1-13	2.98E-10	
	IA1-17	9.00E-11	
	IA1-19	5.17E-09	
	IBL1-05	1.69E-11	
	II1-10	3.89E-10	
	II1-14	9.53E-08	
	II1-15	1.93E-10	
	II1-27	2.08E-09	
	II2-11	3.09E-07	
	II2-23	7.43E-09	
	II2-32	5.39E-10	
	II2-36	1.13E-07	
	II2-37	2.40E-10	
	II2-49	2.46E-09	
	IIE1-14	3.31E-09	
	IIE1-27	2.36E-11	
	IIE2-11	1.14E-08	
	IIE2-23	1.37E-10	
	IIE2-36	1.44E-08	
	IIE2-49	3.28E-11	
	3B-22	1.04E-10	

Table 3-3

QUAD CITIES LEVEL 2 CET SEQUENCE ASSIGNMENT TO APB

Accident Progression Bin	QC Level 2 Sequence[1]		Bin Total (per year)
	Sequence	Frequency (Per year)	
APB #7 (Class 7g)	3B-24	3.89E-10	1.24E-07
	3C-21	1.08E-09	
	IA1-21	3.35E-09	
	IA1-22	1.19E-08	
	IA1-24	3.25E-09	
	IA1-25	2.38E-08	
	IA2-19	1.26E-09	
	IA2-20	2.30E-08	
	IA2-22	6.28E-09	
	IA2-23	4.56E-08	
	IBE1-24	9.86E-11	
	IBE2-22	5.63E-12	
	IBL1-24	1.13E-09	
	IBL1-25	8.84E-11	
	IBL2-22	8.27E-11	
	ID1-22	1.79E-11	
	ID1-24	1.53E-11	
	ID1-25	1.36E-11	
	II2-07	1.61E-09	
	II2-19	1.01E-11	
	IIE2-07	1.67E-11	
	IVA2-07	6.41E-10	
	IVA2-32	6.41E-10	
APB #8[2] (No Category)	--	--	--

Table 3-3

QUAD CITIES LEVEL 2 CET SEQUENCE ASSIGNMENT TO APB

Accident Progression Bin	QC Level 2 Sequence[1]		Bin Total (per year)
	Sequence	Frequency (Per year)	
APB #9 (Category 7h)	3B-01	3.88E-11	4.07E-08
	3B-04	1.43E-09	
	IA1-01	3.35E-09	
	IA1-04	3.37E-08	
	IA1-09	1.12E-09	
	IBE1-01	3 43E-11	
	IBE1-04	5.75E-11	
	IBL1-01	3.29E-10	
	IBL1-04	5.82E-10	
	IBL2-01	2.75E-11	
	IBL2-04	3.96E-11	
	IC1-01	6.78E-12	
Category 7 Total	--	--	1.58E-6

(1) Only sequences with non-negligible frequencies (i e., above the truncation limit) are sorted into the APBs. Sequences associated with Accident Class V are not included in this table since containment bypass scenarios are accounted for in EPRI Category 8. Sequence data compiled from QC-L2-RESULTS-R1.xls

(2) No Quad Cities Level 2 end states matched the definition for APB #8

Table 3-4

SUMMARY OF QUAD CITIES BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	No Containment Failure: Accident sequences in which the containment remains intact and is initially isolated Only affected by ILRT leak testing frequency due to the incorporation of categories 3a and 3b.	Per NEI Interim Guidance: *[Total QC "OK" release category frequency] – [Frequency EPRI Categories 3a and 3b]* *4 97E-7/yr – 3 26E-8/yr - 3.26E-9yr = 4.61E-7/yr*	4.61E-7
2	Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails during the accident progression (e.g , due to failures-to-close of large containment isolation valves initiated by support system failures, or random or common cause failures) Not affected by ILRT leak testing frequency.	[QC containment isolation failure probability] X [Total CDF – CDF of Class II - CDF of Class IIID - CDF of Class IV - CDF of Class V] [2.90E-3] X [2 18E-6/yr – 6.45E-7/yr - 1.19E-8/yr - 1.70E-7/yr - 1.75E-8/yr] = 3.88E-9/yr	3.88E-9
3a	Small Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[QC CDF for accidents not involving containment failure/bypass] x [2.7E-2]* *[2.18E-6/yr – 3.14E-7/yr - 6 45E-7/yr - 1.75E-8/yr] x [2.70E-02] = 3 26E-8/yr*	3 26E-8
3b	Large Pre-Existing Failures: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure or liner that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).	Per NEI Interim Guidance: *[QC CDF for accidents not involving containment failure/bypass] x [2.7E-3]* *[2.18E-6/yr – 3 14E-7/yr - 6 45E-7/yr - 1.75E-8/yr] x [2.70E-03] = 3.26E-9/yr*	3 26E-9
4	Type B Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency)	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A

Table 3-4

SUMMARY OF QUAD CITIES BASELINE RELEASE FREQUENCIES AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
5	Type C Failures: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance. N/A (not affected by ILRT frequency)	N/A
6	Other Containment Isolation System Failure: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	N/A
7	Containment Failure Due to Severe Accident: Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.	Assignment of QC Level 2 sequences to NUREG/CR-4551 APBs See Table 3-3 7a 2.69E-9/yr 7b 1.85E-9/yr 7c 1.05E-7/yr 7d 4.90E-7/yr 7e 1.33E-9/yr 7f 8.15E-7/yr 7g 1.24E-7/yr 7h 4.07E-8/yr	1.58E-6
8	Containment Bypass Accidents. Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., Break Outside Containment LOCA, or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	*[Total QC Containment Bypass release frequency]*	1.75E-8
		TOTAL:	2.10E-6 (1)

(1) Accurate to within a few percent of the [18] total release (2.18E-6/yr) due to rounding and the calculational approach

3.2 CONTAINMENT LEAKAGE RATES (STEP 2)

The second step of the NEI Interim Guidance is to define the containment leakage rates for EPRI Categories 3a and 3b. As discussed earlier, EPRI Categories 3a and 3b are accidents with pre-existing containment leakage pathways ("small" and "large", respectively) that would only be identifiable from an ILRT.

The NEI Interim Guidance recommends containment leakage rates of $10L_a$ and $35L_a$ for Categories 3a and 3b, respectively. The NEI Interim Guidance describes these two recommended containment leakage rates as "conservative". These values are consistent with previous ILRT frequency extension submittal applications. L_a is the plant Technical Specification maximum allowable containment leak rate; for Quad Cities L_a is 1.0% of containment air weight per day (per Quad Cities Technical Specifications).

The NEI recommended values of $10L_a$ and $35L_a$ are used as is in this analysis to characterize the containment leakage rates for Categories 3a and 3b.

By definition, the containment leakage rate for Category 1 (i.e., accidents with containment leakage at or below maximum allowable Technical Specification leakage) is $1.0L_a$.

3.3 BASELINE POPULATION DOSE RATE ESTIMATES (STEPS 3-4)

The third and fourth steps of the NEI Interim Guidance are to estimate the baseline population dose (person-rem) for each EPRI category and to calculate the dose rate (person-rem/year) by multiplying the category frequencies by the estimated dose.

3.3.1 Population Dose Estimates (Step 3)

The NEI Interim Guidance recommends two options for calculating population dose for the EPRI categories:

- Use of NUREG-1150 dose calculations
- Use of plant-specific dose calculations

The NUREG-1150 [14] dose calculations were used in the EPRI TR-104285 study, as discussed previously in Section 2.1. The use of generic dose information for NUREG-1150 is recommended by NEI to make the ILRT risk assessment methodology more readily usable for plants that do not have a Level 3 PSA. As Quad Cities does not have a Level 3 PSA or associated plant-specific dose calculations, this ILRT risk assessment employs NUREG-1150 dose results calculated using the MACCS2 (MELCOR Accident Consequence Code System) consequence code; specifically, the doses for the Peach Bottom NUREG-1150 study (as documented in supporting report NUREG/CR-4551) are used. The following discussion summarizes the population dose calculation and results.

Peach Bottom NUREG-1150 Study Population Dose

The population dose is calculated by using data provided in NUREG/CR-4551 for Peach Bottom and adjusting the results for Quad Cities. Each accident sequence was associated with an applicable collapsed Accident Progression Bin (APB) from NUREG/CR-4551. The definitions of the ten collapsed APBs are reproduced in Table 3-1.

Table 3-5 summarizes the calculated population dose associated with each APB from NUREG/CR-4551, Vol. 4, for Peach Bottom including the fraction of the population dose within 50 miles contributed by each APB and the frequency of release.

Adjustment of NUREG-4551 Doses to Quad Cities

As discussed in Section 2.4.3, the Peach Bottom NUREG/CR-4551 ex-plant consequence results are used as input to determine the population dose estimates of this risk assessment. The NUREG/CR-4551 consequences summarized in Table 3-5 are adjusted for use in this analysis to account for differences in the following parameters between NUREG-1150 analysis and the Quad Cities plant to obtain realistic estimates for Quad Cities:

- Population
- Reactor Power Level
- Technical Specification Allowed Containment Leakage Rate

Population Adjustment

As discussed in Section 2.4.3, the 50-mile radius Peach Bottom population used in the NUREG/CR-4551 consequence calculations is estimated at 3.2E+6 persons, whereas the year 2000 population within the 50-mile radius of Quad Cities is estimated at 7.0E+5 persons. This difference in population results in the adjustment factor to be applied to the NUREG/CR-4551 APB doses of 0.22.

Reactor Power Level Adjustment

As discussed in Section 2.4.3, the reactor power level used in the NUREG/CR-4551 Peach Bottom consequence calculations is 3293 MWth, whereas the Quad Cities Extended Power Uprate full power level is 2957 MWth. This difference in reactor power level results in the following adjustment factor to be applied to the NUREG/CR-4551 APB doses: 0.90.

Containment Leakage Rate Adjustment

As discussed in Section 2.4.3, the containment leakage rate used in the NUREG/CR-4551 consequence calculations for core damage accidents with the containment intact is 0.5 Vol^{PB} % over 24 hours, whereas the Quad Cities maximum allowable containment leakage per Technical Specifications is 1.0 Vol^{QC} % per day. While use of a leakage rate below the maximum allowable may be reasonable, this analysis assumes that containment leakage is at the maximum allowable Technical Specification value. Additionally, a correction is required to account for differences in containment volumes. The containment volume of Peach Bottom is 2.97E+5 ft^3 while that of Quad Cities is slightly smaller, 2.73E+5 ft^3. These differences result in an adjustment factor of 1.84 to be applied to the NUREG/CR-4551 APB doses.

Population Dose by APB for Quad Cities

Table 3-6 provides the translation of the surrogate analysis (Peach Bottom from NUREG-4551) to the Quad Cities plant and site based on APBs. This translation uses the adjustments to power, population, and containment leak rate to the NUREG/CR-4551 population dose results at 50 miles to obtain the adjusted population dose at 50 miles for Quad Cities for each APB.

Population Dose by EPRI Category for Quad Cities

Using the preceding information, the population dose as a function of EPRI category for the 50-mile radius surrounding Quad Cities is summarized in Table 3-8. The following discussion provides the basis for the assignment of population dose for each EPRI category. Note that all population doses are derived from the scaled dose estimates of the surrogate plant (see Table 3-6).

The dose for EPRI category #1 (core damage accident with isolated and intact containment, i.e., no containment failure) is based on NUREG/CR-4551 APB #8, the APB closest to the definition of an intact containment adjusted for population, power, and technical specification leakage as shown in Table 3-6.

The dose for EPRI Category 2 for core damage accidents with containment isolation failures is based on NUREG/CR-4551 APB #3. This assignment is based on assuming that the containment isolation failure of EPRI Category 2 occurs in the drywell as an unscrubbed release. APB #3 results in the highest dose of all the Peach Bottom "containment failure" APBs (which is indicative of a containment failure with an unscrubbed release).

No separate assignment of NUREG/CR-4551 APBs is made for EPRI Categories 3a and 3b. Instead, per the NEI Interim Guidance, the doses for EPRI Categories #3a and #3b are taken as factors of 10 and 35, respectively, times the population dose of EPRI Category 1.

As EPRI Categories 4, 5, and 6 are not affected by ILRT frequency and not analyzed as part of this risk assessment (per NEI Interim Guidance), no assignment of NUREG/CR-4551 APBs is made for these categories.

The dose for EPRI Category 7 is based on the development of a weighted average person-rem dose representative of the EPRI Category 7 subcategories 7a - 7h. This weighted average approach is acceptable since the total frequency and dose associated with EPRI Category 7 does not change as part of the ILRT extension. Table 3-7 summarizes the dose for subcategories 7a - 7h and the representative Category 7 dose. The representative dose for EPRI Category 7 is 4.42E+5.

Table 3-5

PEACH BOTTOM NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE CALCULATION(1)

APB #	APB Definition	APB Frequency (per year) (2)	APB Fractional Contribution to 50-Mile Radius Total Dose Risk(3)	APB 50-Mile Radius Dose Risk (person-rem/year) (4)	APB 50-Mile Radius Dose (Person-rem) (5)
1	CD, VB, Early CF, WW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means Direct Containment Heating (DCH) is possible).	9.55E-8	0.021	0.166	1.74E+6
2	CD, VB, Early CF, WW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).	4.77E-8	0 0066	0 0521	1.09E+6
3	CD, VB, Early CF, DW Failure, RPV Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).	1.48E-6	0.556	4.39	2.97E+6

Table 3-5

PEACH BOTTOM NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE CALCULATION(1)

APB #	APB Definition	APB Frequency (per year) (2)	APB Fractional Contribution to 50-Mile Radius Total Dose Risk(3)	APB 50-Mile Radius Dose Risk (person-rem/year) (4)	APB 50-Mile Radius Dose (Person-rem) (5)
4	CD, VB, Early CF, DW Failure, RPV Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).	7.94E-7	0.226	1.79	2.25E+6
5	CD, VB, Late CF, WW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the wetwell (i.e., after vessel breach during Molten Core-Concrete Interaction (MCCI)) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.	1.30E-8	0.0022	0.0174	1.34E+6
6	CD, VB, Late CF, DW Failure, N/A Core Damage occurs followed by vessel breach. The containment fails late in the drywell (i.e., after vessel breach during MCCI) and the RPV pressure is not important since, even if DCH occurred, it did not fail containment at the time it occurred.	2.04E-7	0.059	0.466	2.28E+6

Table 3-5

PEACH BOTTOM NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE CALCULATION(1)

APB #	APB Definition	APB Frequency (per year) (2)	APB Fractional Contribution to 50-Mile Radius Total Dose Risk(3)	APB 50-Mile Radius Dose Risk (person-rem/year) (4)	APB 50-Mile Radius Dose (Person-rem) (5)
7	CD, VB, No CF, Vent, N/A Core Damage occurs followed by vessel breach. The containment never structurally fails, but is vented sometime during the accident progression. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH does not significantly affect the source term as the containment does not fail and the vent limits its effect.	4.77E-7	0.118	0.932	1.95E+6
8	CD, VB, No CF, N/A, N/A Core damage occurs followed by vessel breach. The containment never fails structurally (characteristic 4 is N/A) and is not vented. RPV pressure is not important (characteristic 5 is N/A) since, even if it occurred, DCH did not fail containment. Some nominal leakage from the containment exists and is accounted for in the analysis so that while the risk will be small it is not completely negligible.	7.99E-7	0.0005	3.95E-3	4.94E+3
9	CD, No VB, N/A, N/A, N/A Core damage occurs but is arrested in time to prevent vessel breach. There are no releases associated with vessel breach or MCCI. It must be remembered, however, that the containment can fail due to overpressure or venting even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment.	3.86E-7	0.01	0.079	2.05E+5

Table 3-5

PEACH BOTTOM NUREG/CR-4551 50-MILE RADIUS POPULATION DOSE CALCULATION[1]

APB #	APB Definition	APB Frequency (per year) [2]	APB Fractional Contribution to 50-Mile Radius Total Dose Risk[3]	APB 50-Mile Radius Dose Risk (person-rem/year) [4]	APB 50-Mile Radius Dose (Person-rem) [5]
10	No CD, N/A, N/A, N/A, N/A Core damage did not occur. No in-vessel or ex-vessel release occurs. The containment may fail on overpressure or be vented. The RPV may be at high or low pressure depending on the progression characteristics The risk associated with this bin is negligible.	4.34E-8	0	0	0
	Total:	4.34E-6	1.0	7.9	--

(1) This table is presented in the form of a calculation because NUREG/CR-4551 does not document dose results as a function of accident progression bin (APB); as such, the dose results as a function of APB must be back calculated from documented APB frequencies and APB dose risk results.

(2) The total (i.e., internal accident sequences) CDF of 4.34E-6/yr and the CDF subtotals by APB are taken from Figure 2.5-6 of NUREG/CR-4551 Vol. 4, Rev. 1, Part I.

(3) The individual APB contributions to the total (i.e., internal accident sequences) 50-mile radius dose rate are taken from Table 5.2-3 of NUREG/CR-4551 Vol. 4, Rev. 1, Part I.

(4) The APB 50-mile dose risk is calculated by multiplying the individual APB dose risk fractional contributions (column 4) by the total 50-mile radius dose risk of 7.9 person-rem/yr (taken from Table 5.1-1 of NUREG/CR-4551 Vol. 4, Rev. 1, Part I).

(5) The individual APB doses are calculated by dividing the individual APB dose risk (Column 5) by the APB frequencies (Column 3).

Table 3-6

QUAD CITIES POPULATION DOSE BY APB:

ADJUSTED PEACH BOTTOM NUREG/CR-4551
50-MILE RADIUS POPULATION DOSES

APB #	Peach Bottom 50-Mile Radius Dose (Person-rem) [1]	Population Adjustment Factor	Reactor Power Adjustment Factor	Containment Leak Rate Adjustment Factor	Quad Cities Population Dose Adjusted 50-Mile Radius Dose (Person-rem)
1	1.74E+06	0.22	0.90	n/a	3 45E+05
2	1.09E+06	0.22	0.90	n/a	2.16E+05
3	2.97E+06	0 22	0.90	n/a	5.88E+05
4	2.25E+06	0 22	0.90	n/a	4 46E+05
5	1.34E+06	0 22	0.90	n/a	2.65E+05
6	2.28E+06	0 22	0.90	n/a	4.51E+05
7	1.95E+06	0.22	0.90	n/a	3 86E+05
8	4.94E+03	0 22	0.90	1.84	1.80E+03
9	2.05E+05	0.22	0.90	n/a	4.06E+04
10	0	0 22	0.90	n/a	0

(1) From Table 3-5

The dose for the containment bypass category, EPRI Category 8, is based on NUREG/CR-4551 APB #3. APB #3 results in the highest dose of all the NUREG/CR-4551 "containment failure" APBs, indicative of containment bypass scenarios.

3.3.2 Baseline Population Dose Rate Estimates (Step 4)

The baseline dose rates per EPRI accident category are calculated by multiplying the population dose estimates from Table 3-8 by the frequencies summarized in Table 3-4 The resulting baseline population dose rates by EPRI category are summarized in Table 3-9. As the conditional containment pre-existing leakage probabilities for EPRI Categories 3a and 3b are reflective of a 3-per-10 year ILRT frequency (refer to Section 3.1), the baseline results shown in Table 3-9 are indicative of a 3-per-10 year ILRT surveillance frequency.

3.4 IMPACT OF PROPOSED ILRT INTERVAL (STEPS 5-9)

Steps 5 through 9 of the NEI Interim Guidance assess the impact on plant risk due to the new ILRT surveillance interval in the following ways:

- Determine change in probability of detectable leakage (Step 5)
- Determine population dose rate for new ILRT interval (Step 6)
- Determine change in dose rate due to new ILRT interval (Step 7)
- Determine change in LERF risk measure due to new ILRT interval (Step 8)
- Determine change in CCFP due to new ILRT interval (Step 9)

Table 3-7

QUAD CITIES EPRI CATEGORY 7 POPULATION DOSE RATE

EPRI Category (APB #)	QC Release Frequency per year[1]	QC Population Dose (50 miles) Person-Rem [2]	Population Dose Risk (50 mile) Person-Rem/yr[3]
7a (APB #1)	2.69E-9	3.45E+5	9.28E-4
7b (APB #2)	1.85E-9	2.16E+5	4.00E-4
7c (APB #3)	1.05E-7	5.88E+5	6.17E-2
7d (APB #4)	4.90E-7	4.46E+5	2.19E-1
7e (APB #5)	1.33E-9	2.65E+5	3.52E-4
7f (APB #6)	8.15E-7	4.51E+5	3.68E-1
7g (APB #7)	1.24E-7	3.86E+5	4.79E-2
7h (APB #9)	4.07E-8	4.06E+4	1.65E-3
Category 7 Total	1.58E-6	4.42E+5[4]	6.99E-1

Notes:

(1) Table 3-3

(2) Table 3-6

(3) Obtained by multiplying the release frequency (column 2) by the population dose (column 3)

(4) Weighted average population dose for Category 7 obtained by dividing the total population dose risk (0 699 Person-Rem/yr) by the total release frequency (1.58E-6/yr).

Table 3-8

EPRI Category	Category Description	QC Person-Rem Within 50 miles
1	No Containment Failure[1]	1.80E+3
2	Containment Isolation System Failure[2]	5.88E+5
3a	Small Pre-Existing Failures[3]	1.80E+4
3b	Large Pre-Existing Failure[4]	6.30E+4
4	Type B Failures (LLRT)[5]	n/a
5	Type C Failures (LLRT)[5]	n/a
6	Other Containment Isolation System Failure[5]	n/a
7	Containment Failure Due to Severe Accident[6]	4.42E+5
8	Containment Bypass Accidents[2]	5.88E+5

Notes

(1) Based on APB #8 of Table 3-6

(2) Based on APB #3 of Table 3-6

(3) Factor of 10 times EPRI Category 1

(4) Factor of 35 times EPRI Category 1

(5) Not analyzed since not affected by ILRT frequency

(6) Weighted average of subcategories 7a-7h of Table 3-7

Table 3-9

QUAD CITIES DOSE RATE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50 MILES
(Base Line 3/10 year ILRT)

EPRI Category	Category Description	Person-Rem Within 50 miles[6]	Baseline Frequency (per year)[7]	Dose Rate (Person-Rem/yr)
1	No Containment Failure[1]	1.80E+3	4.61E-7	8.30E-4
2	Containment Isolation System Failure[2]	5.88E+5	3.88E-9	2.28E-3
3a	Small Pre-Existing Failures[3]	1.80E+4	3.26E-8	5.87E-4
3b	Large Pre-Existing Failures[3]	6.30E+4	3.26E-9	2.05E-4
4	Type B Failures (LLRT)	n/a	n/a	n/a
5	Type C Failures (LLRT)	n/a	n/a	n/a
6	Other Containment Isolation System Failure	n/a	n/a	n/a
7	Containment Failure Due to Severe Accident[4]	4.42E+5	1.58E-6	6.99E-1
8	Containment Bypass Accidents[5]	5.88E+5	1.75E-8	1.03E-2
	Total		2.86E-5[8]	7.13E-1

Notes to Table 3-9

(1) The population dose associated with the Technical Specification Leakage is based on scaling the population data, the power level, and allowable Technical Specification leakage compared to the NUREG/CR-4551 reference plant. The release for this EPRI category is assigned from APB #8 from Table 3-6.

(2) EPRI Category 2 (Containment Isolation failures) may include drywell isolation failures. Therefore, the release associated with this category is assigned to be equivalent to the release associated with APB #3 from Table 3-6.

(3) Dose estimates for categories 3a and 3b, per the NEI Interim Guidance, are calculated as 10xCategory 1 dose and 35xCategory 1 dose, respectively.

(4) Dose estimate for category 7 is the weighted average of subcategories 7a-7h of Table 3-7.

(5) EPRI Category 8 sequences involve containment bypass failures; as a result, the person-rem dose is not based on normal containment leakage. The releases for this category are assumed to result in a direct path to the environment, and as such, are assigned to be equivalent to the highest release category from NUREG/CR-4551. APB #3 from Table 3-6 is therefore used.

(6) Table 3-8.

(7) Table 3-4.

(8) Within a few percent of total release frequency of 2.76E-5/yr [18]. Slight differences due to calculational approach and round off.

3.4.1 Change in Probability of Detectable Leakage (Step 5)

Step 5 of the NEI Interim Guidance is the calculation of the change in probability of leakage detectable only by ILRT (and associated re-calculation of the frequencies of the impacted EPRI categories). Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rates are assumed not to change; however, the probability of pre-existing leakage detectable only by ILRT does increase.

Per the NEI Interim Guidance, the calculation of the change in the probability of a pre-existing ILRT-detectable containment leakage is based on the relationship that relaxation of the ILRT interval results in increasing the average time that a pre-existing leak would exist undetected. Using the standby failure rate statistical model, the average time that a pre-existing containment leak would exist undetected is one-half the surveillance interval. For example, if the ILRT frequency is 1-per-10 years, then the average time that a leak would be undetected is 60 months (surveillance interval of 120 months divided by 2). The impact on the leakage probability due to the ILRT interval extension is then calculated by applying a multiplier determined by the ratio of the average times of undetection for the two ILRT interval cases.

As discussed earlier in Section 3.1, the conditional probability of a pre-existing ILRT-detectable containment leakage is divided into two categories. The calculated pre-existing ILRT-detectable leakage probabilities are reflective of a 3-per-10 year ILRT frequency and are as follows:

- "Small" pre-existing leakage (EPRI Category 3a): 2.70E-2
- "Large" pre-existing leakage (EPRI Category 3b): 2.70E-3

Since the latter half of the 1990's, the Quad Cities plant has been operating under a 1-per-10 year ILRT testing frequency consistent with the performance-based Option B of

10 CFR Part 50, Appendix J. [16] The baseline[1] leakage probabilities first need to be adjusted to reflect the current 1-per-10 year Quad Cities ILRT testing frequency, as follows:

- "Small" : 2.70E-2 x [(120 months/2) / (36 months/2)] = 9.00E-2
- "Large" : 2.70E-3 x [(120 months/2) / (36 months/2)] = 9.00E-3

Note that a nominal 36 month interval (i.e., as opposed to 40 months, 120/3) is used in the above adjustment calculation to reflect the 3-per-10 year ILRT frequency. This is consistent with operational practicalities and the NEI Interim Guidance.

Similarly, the pre-existing ILRT-detectable leakage probabilities for the 1-per-15 year ILRT frequency currently being pursued by Quad Cities (and the subject of this risk assessment) are calculated as follows:

- "Small" : 9.00E-2 x [(180 months/2) / (120 months/2)] = 1.35E-1
- "Large" : 9.00E-3 x [(180 months/2) / (120 months/2)] = 1.35E-2

Given the above adjusted leakage probabilities, the impacted frequencies of the EPRI categories are summarized below (refer to Table 3-4 for details regarding frequency calculations for the individual EPRI categories):

EPRI Category	EPRI Category Frequency as a Function of ILRT Interval		
	Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	4.61E-7	3.78E-7	3.18E-7
3a	3.26E-8	1.09E-7	1.63E-7
3b	3.26E-9	1.09E-8	1.63E-8

(1) The baseline case uses data characteristic of the 3/10 year ILRT frequency of testing.

Note that, per the definition of the EPRI categories, only the frequencies of Categories 1, 3a, and 3b are impacted by changes in ILRT testing frequencies.

3.4.2 Population Dose Rate for New ILRT Interval (Step 6)

The dose rates per EPRI accident category as a function of ILRT interval are summarized in Table 3-10.

3.4.3 Change in Population Dose Rate Due to New ILRT Interval (Step 7)

As can be seen from the dose rate results summarized in Table 3-10, the calculated total dose rate increases imperceptibly (0.001 person-rem/yr) from the current Quad Cities 1-per-10 year ILRT interval amount of 7.15E-1 person-rem/year to the proposed 1-per-15 year ILRT interval dose rate of 7.16E-1 person-rem/year.

Per the NEI Interim Guidance, the change in percentage contribution to total dose rate attributable to EPRI Categories 3a and 3b is also investigated here. Using the results summarized in Table 3-10, for the current Quad Cities 1-per-10 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b is shown to be very minor:

$$[(1.96E\text{-}3 + 6.84E\text{-}4) / 7.15E\text{-}1] \times 100 = 0.37\%$$

For the proposed 1-per-15 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b increases slightly but remains very minor:

$$[(2.93E\text{-}3 + 1.03E\text{-}3) / 7.16E\text{-}1] \times 100 = 0.55\%$$

Table 3-10

BASELINE DOSE RATE ESTIMATES BY EPRI ACCIDENT CATEGORY FOR POPULATION WITHIN 50 MILES

EPRI Category	Category Description	Dose Rate as a Function of ILRT Interval (Person-Rem/Yr)		
		Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	No Containment Failure	8 30E-4	6.80E-4	5.72E-4
2	Containment Isolation System Failure	2.28E-3	2.28E-3	2.28E-3
3a	Small Pre-Existing Failures	5.87E-4	1.96E-3	2.93E-3
3b	Large Pre-Existing Failures	2.05E-4	6.84E-4	1.03E-3
4	Type B Failures (LLRT)	N/A	N/A	N/A
5	Type C Failures (LLRT)	N/A	N/A	N/A
6	Other Containment Isolation System Failure	N/A	N/A	N/A
7	Containment Failure Due to Severe Accident	6 99E-1	6 99E-1	6.99E-1
8	Containment Bypass Accidents	1.03E-2	1.03E-2	1.03E-2
	TOTAL	7.13E-1	7.15E-1	7.16E-1

3.4.4 Change in LERF Due to New ILRT Interval (Step 8)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would not result in a radionuclide release from an intact containment could in fact result in a release due to the increase in probability of failure to detect a pre-existing leak. Per the NEI Interim Guidance, only Category 3b sequences have the potential to result in large releases if a pre-existing leak were present. As such, the change in LERF (Large Early Release Frequency) is determined by the change in the frequency of Category 3b.

Category 1 accidents are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Similarly, Category 3a is a "small" pre-existing leak. Other accident categories such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval. Late releases are excluded regardless of the size of the leak because late releases are, by definition, not LERF contributors.

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)

= 1.63E-8/yr – 1.09E-8/yr

= 5.4E-9/yr

This delta LERF of 5.4E-9/yr falls into Region III, Very Small Change in Risk, of the acceptance guidelines in NRC Regulatory Guide 1.174. Therefore, increasing the ILRT interval at Quad Cities from the currently allowed 1-per-10 years to 1-per-15 years represents a very small change in risk, and is an acceptable plant change from a risk perspective.

3.4.5 Impact on Conditional Containment Failure Probability (Step 9)

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis. In this assessment, based on the NEI Interim Guidance, CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state (EPRI Category 1) and small failures (EPRI Category 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Consequently, the change in CCFP can be calculated by the following equation:

$$CCFP_{\%} = [1 - (\text{Intact Containment Frequency} / \text{Total CDF})] \times 100\%, \text{ or}$$
$$= [1 - ((\#1 \text{ Frequency} + \#3a \text{ Frequency}) / \text{CDF})] \times 100\%$$

For the 10-year interval:

$$CCFP_{10} = [1 - ((3.78E\text{-}7 + 1.09E\text{-}7) / 2.18E\text{-}6)] \times 100\%$$
$$= 77.7\%$$

For a 15-year interval:

$$CCFP_{15} = [1 - ((3.18E\text{-}7 + 1.63E\text{-}7) / 2.18E\text{-}6)] \times 100\%$$
$$= 78.0\%$$

Therefore, the change in the conditional containment failure probability is:

$$\Delta\, CCFP = CCFP_{15} - CCFP_{10} = 0.3\%$$

This change in CCFP of less than 1% is insignificant from a risk perspective.

Section 4
RESULTS SUMMARY

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 18, 20] have led to the quantitative results summarized in this section. These results demonstrate a very small impact on risk associated with the one time extension of the ILRT test interval to 15 years.

The analysis performed examined Quad Cities specific accident sequences in which the containment remains intact or the containment is impaired. The accidents are analyzed and the results are displayed according to the eight (8) EPRI accident categories defined in Reference [2]:

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

The quantitative results are summarized in Table 4-1. The key results to this risk assessment are those for the ten year interval (current Quad Cities condition) and the fifteen year interval (proposed change). The 3-per-10 year ILRT is a baseline starting point for this risk assessment given that the pre-existing containment leakage probabilities (estimated based on industry experience - - refer to Section 3.1) are reflective of the 3-per-10 year ILRT testing.

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

- Increasing the current 10 year ILRT interval to 15 years results in an insignificant increase in total population dose rate of 0.2 percentage points.
- The increase in the LERF risk measure is also insignificant, a 5.4E-9/yr increase. This LERF increase is categorized as a "very small" increase per NRC Reg. Guide 1.174.
- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.3 percentage points.

Table 4-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	1.80E+3	4.61E-7	8.30E-4	3.78E-7	6.80E-4	3.18E-7	5.72E-4
2	5 88E+5	3.88E-9	2 28E-3	3.88E-9	2.28E-3	3.88E-9	2.28E-3
3a	1.80E+4	3.26E-8	5 87E-4	1.09E-7	1.96E-3	1.63E-7	2.93E-3
3b	6.30E+4	3.26E-9	2.05E-4	1.09E-8	6.84E-4	1.63E-8	1.03E-3
4	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6	N/A	N/A	N/A	N/A	N/A	N/A	N/A
7	4.42E+5	1.58E-6	6.99E-1	1.58E-6	6 99E-1	1.58E-6	6 99E-1
8	5.88E+5	1.75E-8	1.03E-2	1.75E-8	1 03E-2	1.75E-8	1.03E-2
TOTALS:		2.10E-6[4]	7.13E-1	2.10E-6[4]	7.15E-1	2.10E-6[4]	7.16E-1
Increase in Dose Rate [1]					0.002		0.001
Increase in LERF [2]				7 60E-9		5.4E-9	
Increase in CCFP (%) [3]				0.3%		0.3%	

Notes to Table 4-1:

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT is calculated as: total dose rate for 1-per-15 year ILRT, minus total dose rate for 1-per-10 year ILRT. For each case, the dose rate increase is insignificant.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.4 of the report, the change in LERF is determined by the change in the accident frequency of EPRI Category 3b. For example, the increase in LERF for the proposed 1-per-15 ILRT is calculated as: 3b frequency for 1-per-15 year ILRT, 1.63E-8/yr, minus 3b frequency for 1-per-10 year ILRT, 1.09E-8/yr, equals 5.4E-9/yr.

(3) As discussed in Section 3.4.5, the conditional containment failure probability (CCFP) is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category \#1 Frequency} + \text{Category \#3a Frequency}) / \text{CDF})] \times 100\%$$

(4) Due to the NEI methodology and round off, the total frequency of all severe accidents is slightly less than the QC reported CDF (approximately 4%).

Section 5

CONCLUSIONS

5.1 QUANTITATIVE CONCLUSIONS

The conclusions from the risk assessment of the one time ILRT extension can be characterized by the risk metrics used in previously approved ILRT test interval extensions. These include:

- Change in LERF
- Change in conditional containment failure probability
- Change in population dose rate

5.1.1 LERF

Based on the results from Sections 3 and 4, the main conclusion regarding the impact on plant risk associated with extending the Type A ILRT test frequency from ten years to fifteen years is:

> 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 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once-per-ten years to once-per-fifteen years (using the change in the EPRI Category 3b frequency per the NEI Interim Guidance) is 5.4E-9/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the Quad Cities ILRT interval from 10 to 15 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

5.1.2 CCFP

The change in conditional containment failure probability (CCFP) is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The ΔCCFP is found to be very small (0.3% increase) and represents a negligible change in the Quad Cities defense-in-depth.

5.1.3 Population Dose Rate

The change in population dose rate is also reported consistent with previously approved ILRT interval extension requests. The change in population dose rate from the current 1/10 year ILRT frequency to 1/15 year frequency is an insignificant 0.001 person-rem/yr increase.

5.2 RISK TRADE-OFF

The performance of an ILRT introduces risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real risk impacts associated with the setup and performance of the ILRT during shutdown operation [8]. While these risks have not been quantified for Quad Cities, it is judged that there is a positive (yet unquantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT extension, there are, in fact, positive safety benefits associated with reducing the risk contribution from shutdown risk configurations.

5.3 EXTERNAL EVENTS IMPACT

External hazards were evaluated in the Quad Cities Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Although the external event hazards in the Quad Cities IPEEE were evaluated to varying levels of conservatism, the results of the Quad Cities IPEEE are nonetheless used in this risk assessment to provide a conservative comparison of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment.

The proposed ILRT interval extension impacts plant risk in a limited way. Specifically, the probability of a pre-existing containment leak being the initial containment failure mode given a core damage accident is potentially higher when the ILRT interval is extended. This impact is manifested in the plant risk profile in a similar manner for both internal events and external events.

The spectrum of external hazards has been evaluated in the Quad Cities IPEEE by screening methods with varying levels of conservatism. Therefore, it is not possible at this time to incorporate realistic quantitative risk assessments of all external event hazards into the ILRT extension assessment. As a result, external events have been evaluated as a sensitivity case to show that the conclusions of this analysis would not be altered if external events were explicitly considered.

The quantitative consideration of external hazards is discussed in more detail in Appendix B of this report. As can be seen from Appendix B, if the external hazard risk results of the Quad Cities IPEEE are included in this assessment (i.e., in addition to internal events), the change in LERF associated with the increase in ILRT interval from 10 years to 15 years will be 6.84E-8yr. This delta LERF falls below the Region III boundary of <1E-7/yr and, therefore, is within the NRC RG 1.174 Region III ("Very Small Changes" in risk).

Therefore, incorporating external event accident sequence results into this analysis does not change the conclusion of this risk assessment (i.e., increasing the Quad Cities ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

5.4 INTERNAL FLOOD IMPACTS

The impact of internal flooding events on this ILRT risk assessment is summarized in this section (refer to Appendix B for further details). The purpose is to assess whether there are any unique insights or important qualitative information associated with the explicit consideration of internal flooding events in the risk assessment results.

The increase in LERF associated with internal flooding events (1.9E-10/yr) was found to be negligible being nearly three orders of magnitude below the NRC RG 1.174 Region III upper bound of 1E-7/yr. Internal flooding events have minimal impact on LERF because most of the internal flooding CDF contributions (91%) result in Class II sequences. Class II sequences are generally not LERF contributors and are therefore not included in the LERF calculation (see Appendix B for details).

The use of the draft internal flood evaluation results are therefore judged acceptable for this evaluation. Minor changes (even an order of magnitude change) to the internal flooding analysis results will not change the conclusion of the ILRT interval extension risk assessment (i.e., increasing the Quad Cities ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

5.5 PREVIOUS ASSESSMENTS

The NRC in NUREG-1493 [5] has previously concluded that:

- 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 findings for Quad Cities confirm the above general findings on a plant specific basis when considering the following: (1) Quad Cities severe accident risk profile, (2) the Quad Cities containment failure modes, and (3) the local population surrounding the Quad Cities site.

Section 6

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[2] Electric Power Research Institute, Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI TR-104285, August 1994.

[3] Letter from A. Petrangelo (NEI) to NEI Administrative Points of Contact, *"Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leak Rate Test Surveillance Intervals"*, November 13, 2001.

[4] U.S. Nuclear Regulatory Commission, *"An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"*, Regulatory Guide 1.174, Revision 1, November 2002.

[5] *Performance-Based Containment Leak-Test Program,* NUREG-1493, September 1995.

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[7] United States Nuclear Regulatory Commission, Indian Point Nuclear Generating Unit No. 3 - Issuance of Amendment Re: Frequency of Performance-Based Leakage Rate Testing (TAC No. MB0178), April 17, 2001.

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[14] U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG -1150, December 1990.

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[16] NRC letter to Quad Cities County Station Issuing Technical Specification Amendment to implement the requirements of 10 CFR 50, Appendix J, Option B for performance-based primary reactor containment leakage testing.

[17] ERIN Engineering and Research, Inc., Identification of Risk Implications Due to Extended Power Uprate at Quad Cities, Doc. #C14670108-4441, August 2001.

[18] "Quad Cities Detailed Level 2 Evaluation", QC PSA-015, Rev. 3, Volumes 1 and 2 dated July, 2002.

[19] Not Used.

[20] Not Used.

[21] Letter from A. Petrangelo (NEI) to NEI Administrative Points of Cont*act, "One-Time Extension of Containment Integrated Leak Rate Test Interval – Additional Information"*, November 30, 2001.

APPENDIX A

POPULATION ESTIMATES

Appendix A

POPULATION ESTIMATES

This appendix includes the population estimates for the following:

- Appendix A.1: 50-Mile Radius Population Data Used to Characterize Peach Bottom Population Dose Calculations in NUREG/CR-4551
- Appendix A.2: 50-Mile Radius Population for Quad Cities

A.1 POPULATION DATA USED TO CHARACTERIZE PEACH BOTTOM POPULATION DOSE CALCULATIONS IN NUREG/CR-4551

Background

NEI Interim Guidance for the ILRT internal extension licensing request includes the option to use NRC Ex-Plant consequences from NUREG-1150 if a plant does not have a plant specific Level 3 PSA. This approach is used for the Quad Cities ILRT analysis.

Analysis

The Population Dose (Person-Rem) calculation for the Mark I surrogate source terms is derived from the NRC's landmark study of reactor risks in NUREG-1150 for the Peach Bottom plant. In order to relate that 50-mile population dose calculation from Peach Bottom to Quad Cities, the population information for both sites is needed to properly scale the calculated dose.

NUREG-1150 does not specify the 50-mile radius population for Peach Bottom. This section derives the population within 50 miles of Peach Bottom used to support the NUREG-1150 risk estimates based upon population estimates for other radial distances.

The following table gives the population within certain distances of the Peach Bottom plant as summarized from the MACCS demographic input based on 1980 Census Tapes (NUREG/CR-4551, Vol. 4, Rev. 1, Part 1, Table 4.2-2).

Distance From Plant (Km)	Distance From Plant (miles)	Population
1.6	1.0	118
4.8	3.0	1822
16.1	10.0	28,647
48.3	30.0	989,356
160.9	100.0	14,849,112
563.3	350.0	68,008,584
1609.3	1000.0	154,828,144

Two methods are used to estimate the population within 50 miles of Peach Bottom:

- Method 1: Assume direct proportion of the population with area
- Method 2: Interpolate between estimates for 30 miles and 100 miles as a function of area.

Method 1

This method assumes a constant population density around the Peach Bottom site, thus calculating the population of one area as a direct proportion of another. This population estimation method is performed for both the Peach Bottom 30-mile radius data point and the 100-mile radius data point.

Using the population density of the 30-mile radius data point produces the following 50-mile radius population estimate:

$$\frac{\pi R_{30}^2}{} = \frac{\pi R_{50}^2}{}$$

$$9.89E{+}5 \qquad Pop_{50}$$

$$Pop_{50} = 9.89E{+}5\ (50^2/30^2) = 2.75E{+}6 \text{ persons}$$

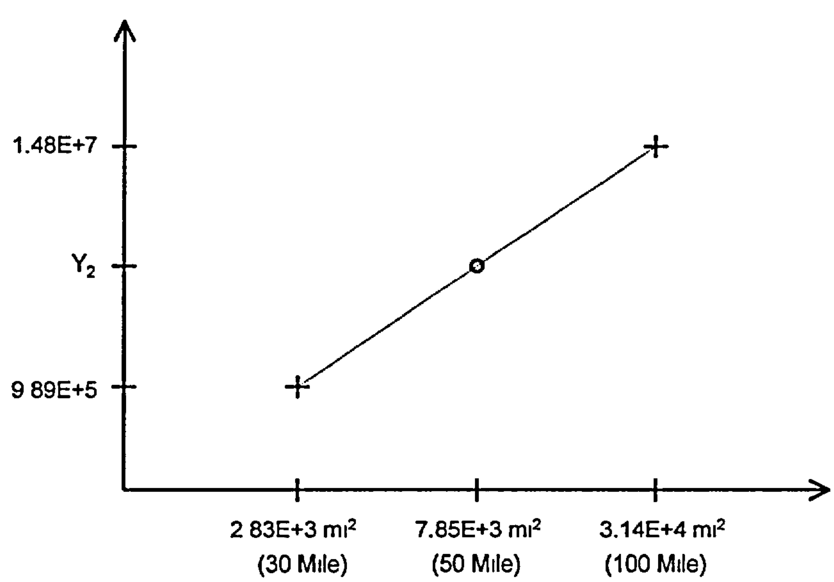
Using the population density of the 100-mile radius data point produces the following 50-mile radius population estimate:

$$Pop_{50} = 1.485E{+}7\ (50^2/100^2) = 3.71E{+}6 \text{ persons}$$

The closeness of these two estimates indicates that the assumptions of a constant population density around the Peach Bottom site is reasonable. The averaged 50-mile radius NUREG-1150 Peach Bottom estimate using the documented 30-mile and 100-mile points is 3.23E+6 persons.

Method 2

This population estimation method is an interpolation assuming a linearly increasing population with distance as a function of area (as shown in the graph below).



$$Y = mx + b \qquad Y_2 - Y_1 = m(X_2 - X_1)$$

$$m = \frac{Y_3 - Y_1}{X_3 - X_1} \qquad Y_1 = 9.89E{+}5$$

$$X_1 = 2.83E{+}3$$

$$Y_2 = 9.89E{+}5 + \frac{1.48E{+}7 - 9.89E{+}5}{3.14E{+}4 - 2.83E{+}3} * (7.85E{+}3 - 2.83E{+}3)$$

Y_2 = 3.42E+6 persons

Summary of Peach Bottom NUREG-1150 50-mile Radius Population

The two population estimation methods yield estimates that are very close. The smaller estimate, 3.2E+6, is chosen for use in this risk assessment since this will lead to a more conservative estimate of the risk at Quad Cities when the person-rem are scaled to the Quad Cities site.

A.2 YEAR 2000 50-MILE RADIUS POPULATION AROUND QUAD CITIES

A calculation of the 50-mile radius population around Quad Cities for the year 2000 was performed in support of this risk assessment.

This calculation uses 2000 Census data, as reported by the US Census Bureau on the web site http://quickfacts.census.gov, along with Iowa and Illinois maps to perform the population estimation.

The site of the Quad Cities Station is in Rock Island County, Illinois. It is on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, and about 3 miles north of Cordova, Illinois. The site is about 20 miles northeast of the Quad Cities (Davenport, Iowa; Rock Island, Moline, and East Moline, Illinois). The location of the site and the 50-mile radius is illustrated in Figure A-1 for Iowa and Figure A-2 for Illinois.

If the entire county falls within the 50-mile radius, based on a review of a map containing a mileage scale and county borders, then the entire population was included in the population estimate. Otherwise, a fraction of the population was counted based on the percentage of the county within the 50-mile radius. The land area within the 50-mile radius was estimated based on visual inspection of the maps and the population of that area was estimated assuming uniform distribution of the population within the county.

Seven counties were completely inside the 50-mile radius (Jackson, Clinton and Scott in Iowa; Carroll, Whiteside, Henry and Rock Island in Illinois). For the other counties, their percentage within the 50-mile radius zone was estimated and then multiplied by their total population based on the 2000 Census data. Since the population density for one of these border counties varied significantly, an exception was made for Lee County, Illinois. The city of Dixon (pop. 15,500) in Lee County lies within the 50-mile zone and accounts for about 43% of the county population (36,000). The remaining Lee County population (20,500) is assumed to be uniformly distributed and was multiplied by the percent of the county that is in the 50-mile zone (35%). As a result, the total Lee County population that falls within the 50-mile zone is estimated as 22,700.

A list of the counties within the 50-mile radius of Quad Cities, along with their total population, the percent land area within the 50-mile radius, and the population within the 50-mile zone is summarized in Tables A-1 and A-2. The total year 2000 population within a 50-mile radius of Quad Cities is estimated at 696,557 persons, rounded up to 7.0E+5.

The Quad Cities UFSAR includes estimates of population centers within the 50-mile radius from Quad Cities. Tables 2.1-1 and 2.1-2 (Rev. 2) from the UFSAR are included for information. It is noted that the 50-mile population estimate for year 2000 is approximately the same as that recorded in UFSAR Table 2.1-2 for the year 1980 (693,769). A review of the Table 2.1-1 for Urban Centers population growth shows a <u>decrease</u> in population from 1980 to 1990 for each Urban Center listed. Section 2.1.3 of the UFSAR summarizes population distribution trends around Quad Cities as follows:

Population growth near the plant since the time of PSAR filing has been slow and generally consistent with the rural population growth rate in the Quad Cities area of about 1% per year maximum. There are no known factors which would change the 1% maximum rural growth rate in the foreseeable future.

Therefore, a 50-mile zone population estimate of 7.0E+5 persons is judged reasonable for the years covered by ILRT evaluation.

Figure A-1

IOWA 50-MILE RADIUS AROUND QUAD CITIES SITE

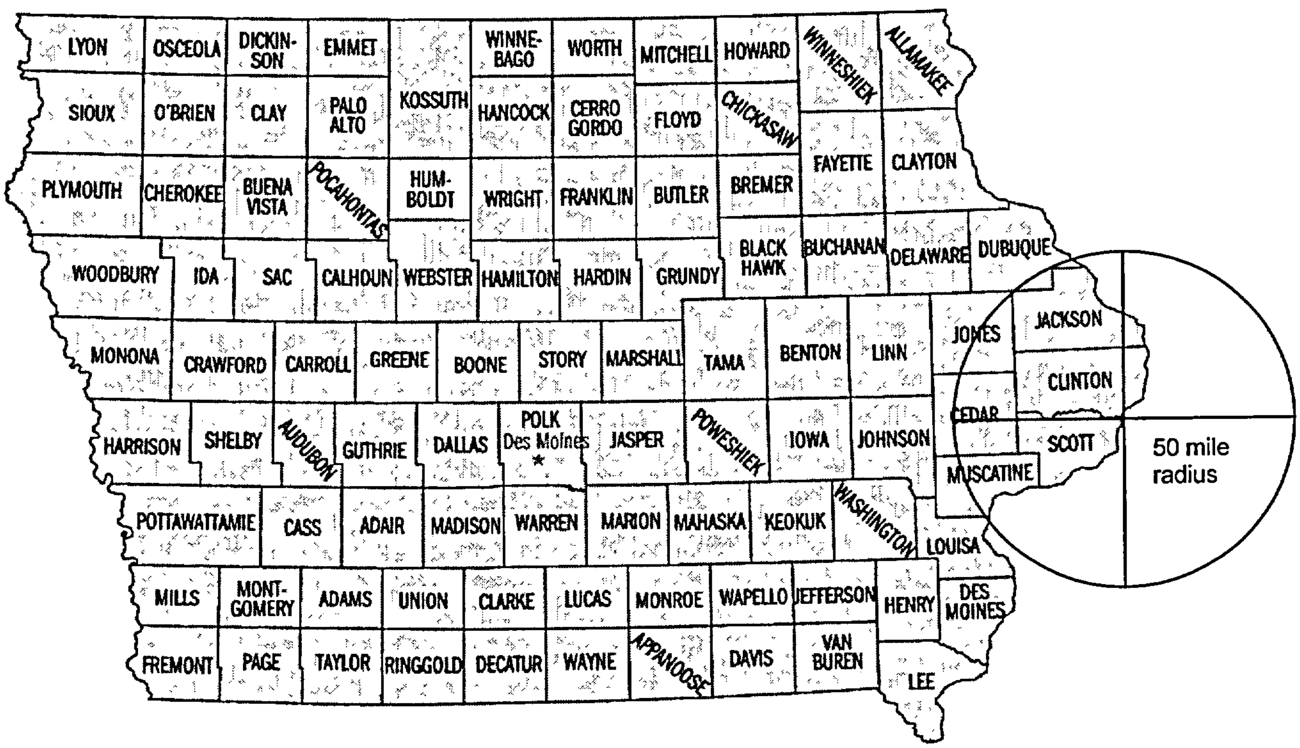


Table A-1

2000 CENSUS POPULATION FOR IOWA COUNTIES
WITHIN 50-MILE RADIUS OF QUAD CITIES

County	2000 Census Population by County[3]	Percent Area of County in 50-Mile Radius[1]	Population within 50-Mile Radius[2]
Cedar	18,187	80	14,550
Clinton	50,149	100	50,149
Dubuque	89,143	10	8,914
Jackson	20,296	100	20,296
Jones	20,221	40	8,088
Muscatine	41,722	60	25,033
Scott	158,668	100	158,668
TOTALS	398,386	—	285,718

(1) Based on visual inspection of Iowa state maps.

(2) County Population multiplied by percentage within the 50-mile zone

(3) Source: http://eire census gov/popest/data/counties/tables/CO-EST2001-07/CO-EST2001-07-19 csv

Figure A-2

ILLINOIS 50-MILE RADIUS AROUND QUAD CITIES SITE

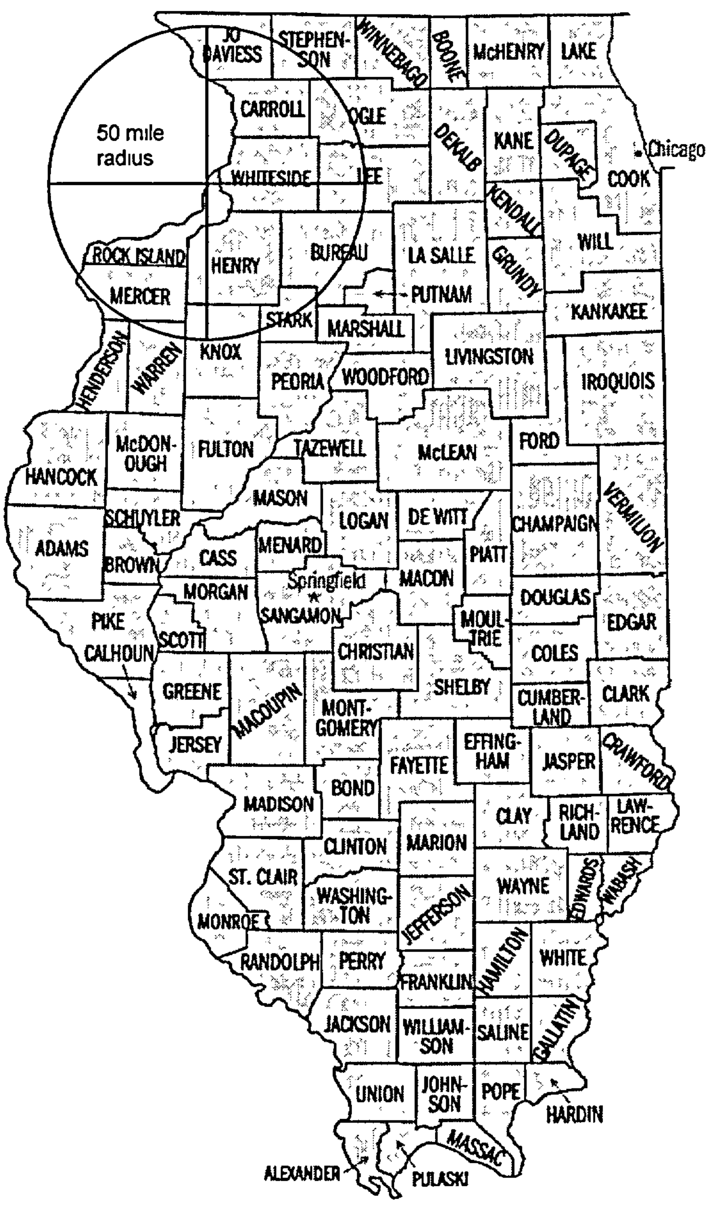


Table A-2

2000 CENSUS POPULATION FOR ILLINOIS COUNTIES
WITHIN 50-MILE RADIUS OF QUAD CITIES

County	2000 Census Population by County(4)	Percent Area of County in 50-Mile Radius(1)	Population within 50-Mile Radius(2)
Bureau	35,503	60	21,302
Carroll	16,674	100	16,674
Henry	51,020	100	51,020
Jo Davies	22,289	75	16,717
Knox	55,836	35	19,543
Lee(3)	36,062	35	22,700(3)
Mercer	16,957	90	15,261
Ogle	51,032	35	17,861
Rock Island	149,374	100	149,374
Stark	6,332	50	3,166
Stephenson	48,979	30	14,694
Warren	18,735	10	1,874
Whiteside	60,653	100	60,653
TOTALS	569,446	—	410,839

(1) Based on visual inspection of Illinois state maps.

(2) County Population multiplied by percentage within 50-mile zone, except when noted.

(3) Population density varied greatly in this region, an exception was made (refer to text).

(4) Source: http://eire.census.gov/popest/data/counties/tables/CO-EST2001-07/CO-EST2001-07-17.csv

QUAD CITIES UFSAR

Table 2.1-1

POPULATION GROWTH — URBAN CENTERS

	Illinois			Iowa		
Year	Rock Island	Moline	East Moline	Davenport	Bettendorf	Clinton
1940 Actual	42,775	34,608	12,359	66,039	3,143	26,270
1950 Actual	48,710	37,397	13,913	74,549	5,132	30,379
1960 Actual	51,863	42,705	16,732	88,981	11,534	33,589
1970 PSAR Estimate	55,000	51,500	21,000	103,500	20,500	36,900
1970 Actual	50,166	46,237	20,956	98,469	22,126	34,719
1980 PSAR Estimate	62,000	60,000	26,000	119,500	35,000	40,600
1980 Actual	47,036	45,709	20,907	103,264	27,376	32,828
1990 Projected	55,967	56,388	28,544	129,676	81,251	42,631
1990 Actual	40,552	43,202	20,147	95,333	28,132	29,201
2000 Projected	40,510	43,150	20,120	101,280	29,890	26,530

QUAD CITIES — UFSAR

Table 2.1-2

SECTORS AND ZONE DESIGNATORS AND CALCULATED 1980 POPULATION DISTRIBUTION WITHIN 50 MILES OF QUAD CITIES STATION

Sector Centerline in Degrees from True North from Facility	22 ½* Sector	Miles from Facility										
		0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50	Sector Total
0 + 360	A	97	24	3	27	945	6,916	945	2,155	5,906	7,893	24,911
22 ½	B	0	8	12	236	1,418	8,502	19,513	6,920	4,086	6,703	47,398
45	C	0	8	16	5	14	4,500	6,880	2,883	2,000	6,899	23,205
67 ½	D	5	5	19	5	14	243	6,966	3,997	7,440	7,458	26,152
90	E	0	5	16	0	41	170	3,432	4,849	19,131	19,498	47,142
112 ½	F	0	0	8	11	14	111	4,040	3,965	4,539	5,547	18,235
135	G	0	5	11	24	19	251	2,361	3,421	4,861	3,458	14,411
157 ½	H	0	5	11	24	19	265	1,964	9,765	19,017	8,054	39,124
180	J	30	73	30	489	19	1,859	9,109	3,582	4,846	6,086	26,123
202 ½	K	0	0	0	407	259	3,238	58,235	11,552	4,265	7,940	85,896
225	L	0	0	19	113	494	337	135,056	81,669	7,707	4,361	229,756
247 ½	M	0	0	30	19	46	157	5,290	6,192	5,524	30,059	47,317
270	N	0	0	14	8	49	311	2,896	3,592	2,699	5,759	15,328
292 ½	P	0	0	68	216	14	399	6,074	2,758	4,033	5,728	19,290
315	Q	0	0	46	27	38	176	957	5,035	6,933	3,295	16,507

Sector Centerline in Degrees from True North from Facility	22 ½* Sector	Miles from Facility										
		0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50	Sector Total
337 ½	R	0	0	24	11	24	2,127	1,554	1,726	2,952	4,556	12,974
Radial Zone Total		132	133	327	1,622	3,427	29,562	265,272	154,061	105,939	133,294	693,769

APPENDIX B

EXTERNAL EVENT AND INTERNAL FLOOD ASSESSMENT

Appendix B

EXTERNAL EVENT AND INTERNAL FLOOD ASSESSMENT

This appendix discusses the external events and internal flood assessment in support of the Quad Cities ILRT interval extension risk assessment.

External hazards were evaluated in the Quad Cities Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program (Generic Letter 88-20 Supplement 4). The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. Although the external event hazards in the Quad Cities IPEEE were evaluated to varying levels of conservatism, the results of the Quad Cities IPEEE are nonetheless used in this risk assessment to provide a conservative comparison of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment.

The Quad Cities internal flood analysis has not yet been included in an approved Quad Cities internal events PSA model. A draft internal events PSA model (2002D) which includes internal flooding is currently under development based upon a draft internal flood evaluation [B-7]. Preliminary results from this draft internal flood evaluation [B-7] are used in this risk assessment to provide a comparison of the impact of internal flooding on the conclusions of this ILRT interval extension risk assessment.

B.1 QUAD CITIES IPEEE INTERNAL FIRES ANALYSIS

The Quad Cities plant risk due to internal fires was updated in 1999 as part of the revised Quad Cities 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 Quad Cities Unit 1 CDF contribution due to internal fires in the unscreened fire areas was calculated at 6.60E-5/yr. The breakdown of the Quad Cities fire risk profile is as follows [B-4]:

• Fire-induced loss of decay heat removal scenarios	80.4%
• Fire-induced loss of inventory control scenarios (RPV at low pressure)	4.3%
• Fire-induced loss of inventory control scenarios (RPV at high pressure)	3.9%
• Other fire-induced scenarios (ATWS)	11.4%

This information is used in Section B.4 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

B.2 QUAD CITIES IPEEE SEISMIC ANALYSIS

The Quad Cities seismic risk analysis was performed as part of the Individual Plant Examination of External Events (IPEEE). Quad Cities performed a seismic margins assessment (SMA) following the guidance of NUREG-1407 and EPRI NP-6041. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No core damage frequency sequences were quantified as part of the IPEEE seismic risk evaluation.

Although probabilistic risk information is not directly available from the Quad Cities SMA IPEEE analysis, Reference [B-1] provides a simple method (called the Simplified Hybrid Method) for obtaining a seismic-induced CDF estimate based on results of an SMA analysis. Reference [B-1] has shown that only the plant HCLPF (High Confidence Low Probability of Failure) seismic capacity is needed in order to estimate the seismic CDF within a precision of approximately a factor of two. The approach is as follows:

<u>Step 1:</u> Determine the plant HCLPF seismic capacity C_{HCLPF} from the SMA analysis

<u>Step 2:</u> Estimate the 10% conditional probability of failure capacity $C_{10\%}$ from:

$$C_{10\%} = F_\beta C_{HCLPF}$$
$$F_\beta = e^{1.044\beta}$$

where 1.044 is the difference between the 10% NEP standard normal variable (-1.282) and the 1% NEP standardized normal variable (-2.326).

Experience gained from high quality seismic PSA studies indicates that the plant damage state fragility determined by rigorous convolution will tend to have β_c values in the range of 0.30 to 0.35 (the plant damage state β_c value is equal to or less than the β_c values for the fragilities of the individual components that dominate the seismic risk). As such, the Simplified Hybrid method recommends:

$$C_{10\%} = 1.4C_{HCLPF}$$

<u>Step 3:</u> Determine hazard exceedance frequency $H_{10\%}$ that corresponds to $C_{10\%}$ from hazard curve.

<u>Step 4:</u> Determine seismic risk PF from:

$$PF = 0.5 H_{10\%}$$

Using the Simplified Hybrid Method, an approximation of the Quad Cities seismic-induced CDF is performed here.

<u>Step 1</u>: If the SMA analysis screens out every component on the Seismic Safe Shutdown Paths at the Review Level Earthquake (RLE), the plant HCLPF is equal to the RLE. Otherwise, the plant HCLPF is determined by the lowest seismic capacity component in the seismic safe shutdown paths. Quad Cities falls into the latter category. The Quad Cities RLE specified in the NRC IPEEE program is 0.30g PGA. A number of equipment items were identified during the QC A-46 and seismic IPEEE analyses to have HCLPF capacities less than the 0.30g PGA RLE[1]. These items were addressed either by plant

[1] Note that the finding of HCLPFs lower than the plant RLE is not an indication of any vulnerability, but is consistent with the IPEEE definitions of RLEs The IPEEE RLEs were defined such that high capacity items would be screened and the lower capacity items would be identified.

improvements or other resolutions. Based on these findings and resolutions, the Quad Cities plant HCLPF was ultimately estimated to be at least 0.24g PGA. [B-5]

<u>Step 2</u>: Using the relationship recommended above, the plant 10% capacity point (C10%) is estimated as 1.4 x 0.24g PGA = 0.34g PGA.

<u>Step 3</u>: The seismic hazard curve for the Quad Cities site, based upon EPRI NP-6395-D, is summarized in tabular form in Table B-1. As can be seen from Table B-1, the seismic hazard frequency associated with the 10% capacity point (0.34g PGA) is approximately 4.8E-6/yr.

<u>Step 4</u>: Using the relationship recommended above, the seismic-induced CDF is approximated as 0.50 x 4.8E-6/yr = 2.4E-6/yr.

The Simplified Hybrid Method only provides an overall seismic-induced CDF estimate and does not provide information as to the breakdown of seismic accident sequence types. A more rigorous analysis (e.g., a seismic PSA, or the Rigorous Hybrid Method referred to in Reference [B-1]) is required for such information. Such an analysis was not performed as part of this ILRT risk assessment. However, a Rigorous Hybrid Method calculation was recently completed for another Exelon BWR plant (Limerick) [B-2]. The results of that study (Case #2 of Reference [B-2]) are used here to provide a reasonable approximation of the breakdown of seismic accident sequence types. They are as follows:

- Seismic-induced loss of decay heat removal scenarios ~35%
- Wide-spread failure of seismic safe shutdown SSCs ~20%
- Seismic-induced ATWS scenarios ~15%
- Other seismic-induced accidents (e.g., SBO, loss of coolant makeup, etc.) ~30%

This information is used in Section B.4 of this appendix to provide quantitative insights into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

B.3 OTHER EXTERNAL HAZARDS

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

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

The Quad Cities 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 Quad Cities 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 Quad Cities IPEEE to be negligible contributors to overall plant risk.

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

Table B-1

QUAD CITIES SITE SEISMIC HAZARD CURVE
- EPRI NP-6395-D[(1)]

Peak Ground Acceleration		EPRI Exceedance Frequency (1/yr, mean)
cm/s^2	g	
5	0.01	3.6E-3
50	0.05	2.3E-4
100	0.10	6.3E-5
250	0.25	7.0E-6
335	0.34	4.8E-6[(2)]
500	0.51	7.7E-7
700	0.71	2.2E-7
1000	1.02	4.7E-8

(1) From Table 3-77 and Figure 3-229 of EPRI NP-6395-D, Appendix E.

(2) Frequency for Quad Cities 10% plant capacity interpolated from EPRI data points.

B.4 IMPACT OF EXTERNAL HAZARD RISK ON LERF

The NEI Interim Guidance calculation of delta LERF performed in Section 3 of this report is re-performed here including, in addition to internal event information, the Quad Cities IPEEE external event risk information discussed in the previous sections.

Per the NEI Interim 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 3.1, 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)]

Based on the previous discussion in Sections B.1 through B.3, the Quad Cities external event initiated CDF is approximately 6.60E-5/yr (internal fires) + 2.40E-6/yr (seismic) = 6.84E-5/yr. In addition, the following external event accident scenarios are excluded from the 3b frequency calculation because they cannot result in a LERF release or independently result in LERF:

- Fire-induced loss of decay heat removal scenarios (5.31E-5/yr)
  0.804 x 6.60E-5/yr = 5.31E-5/yr
- Seismic-induced loss of decay heat removal scenarios (8.40E-7/yr)
  0.35 x 2.40E-6/yr = 8.40E-7/yr
- Wide-spread failure of seismic safe shutdown SSCs (4.80E-7/yr)
  0.20 x 2.40E-6/yr = 4.80E-7/yr

Therefore, the baseline (3-per-10 year) frequency of category 3b due to external events is calculated as (2.70E-03) x [(6.84E-5/yr) – (5.31E-5/yr + 8.40E-7/yr + 4.80E-7/yr)] = 3.78E-8/yr.

Using the relationship described in Section 3.4.1 for the impact on 3b frequency due to increases in the ILRT surveillance interval, the EPRI Category 3b frequency for the 1-per-10 year and 1-per-15 year ILRT intervals are calculated as 1.26E-7/yr and 1.89E-7/yr, respectively. Therefore, the change in the LERF risk measure due to extending the ILRT from 1-per-10 years to 1-per-15 years, including both internal and external hazard risk, is estimated as:

	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase
External Events Contribution	1.26E-7/yr	1.89E-7/yr	6 30E-8/yr
Internal Events Contribution	1.09E-8/yr	1.63E-8/yr	5.43E-9/yr
Combined (Internal + External)	1.37E-7/yr	2.05E-7/yr	6.84E-8/yr

Thus the increase in LERF due to the external events contribution is estimated as 6.30E-8/yr. This is approximately an order of magnitude larger than the LERF increase associated with internal events.

B.5 IMPACT OF INTERNAL FLOOD ON LERF

The Quad Cities internal flood analysis has not yet been included in an approved internal events PSA model. A draft PSA model (2002D) which includes internal flooding is currently under development. Preliminary results from the draft internal flood evaluation [B-7] are summarized in Table B-2.

The NEI Interim Guidance calculation of delta LERF performed in Section 3 of this report is re-performed here for the internal flood contribution.

Per the NEI Interim 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 3.1, 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)]

Based on Table B-2 the Quad Cities internal flood initiated CDF is 4.7E-7/yr. The following internal flood accident scenarios are excluded from the 3b frequency calculation because they cannot result in a LERF release or independently result in LERF:

- Class II loss of decay heat removal scenarios (4.3E-7/yr)[(1)]

Therefore, the baseline (3-per-10 year) frequency of category 3b due to internal flooding events is calculated as (2.70E-03) x [4.7E-7/yr – 4.3E-7/yr] = 1.1E-10/yr.

Using the relationship described in Section 3.4.1 for the impact on 3b frequency due to increases in the ILRT surveillance interval, the EPRI Category 3b frequency for the 1-per-10 year and 1-per-15 year ILRT intervals are calculated as 3.6E-10/yr and 5.5E-10/yr, respectively.

[(1)] Per Table 6.6-2 of [B-6], a small percentage (0.0326%) of Class II releases do contribute to LERF for internal events. This small contribution if applied to internal flooding events however, is negligible and is omitted in the calculation (4.3E-7/yr * 3.26E-4 = 1.40E-10/yr); 4.3E-7/yr - 1.40E-10/yr ≈ 4.3E-7/yr).

Thus the increase in LERF due to internal flooding contribution is estimated as 1.9E-10/yr. This is approximately an order of magnitude less than the LERF increase associated with other internal events.

Therefore, the change in the LERF risk measure due to extending the ILRT from 1-per-10 years to 1-per-15 years, including external events, internal flooding, and other internal events, is estimated as:

	3b Frequency (1-per-10 year ILRT)	3b Frequency (1-per-15 year ILRT)	LERF Increase
External Events Contribution	1.26E-7/yr	1.89E-7/yr	6 30E-8/yr
Internal Flood Contribution	3.6E-10/yr	5.5E-10/yr	1.9E-10/yr
Internal Events Contribution	1.09E-8/yr	1.63E-8/yr	5.43E-9/yr
Combined	1.37E-7/yr	2.06E-7/yr	6.86E-8/yr

B.6 COMPARISON TO RG 1.174 ACCEPTANCE GUIDELINES

NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant. As discussed in Section 2 of this report, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 6.86E-8/yr increase in LERF from extending the Quad Cities ILRT frequency from 1-per-10 years to 1-per-15 years falls into Region III ("Very 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 range of 1E-7 to 1E-6 per reactor year (Region II, "Small Change" in risk), the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr. Although not required in this case (since the delta LERF is less than 1E-7 and falls in Region II), the total LERF is calculated in this analysis for completeness.

Per the Quad Cities internal events Level 2 PRA (2002B), the Quad Cities LERF due to internal event accidents is 2.68E-7/yr [B-6]. The LERF due to external events and internal flooding is estimated here using the Quad Cities conditional LERF probabilities as a function of core damage accident type (refer to Tables B-3 and B-4). As can be seen from Table B-3, the external events LERF is estimated at 2.36E-6/yr. From Table B-4, the internal flooding LERF is estimated at 8.7E-10/yr. Therefore, the total LERF for Quad Cities is estimated at 2.68E-7/yr + 2.36E-6/yr + 8.7E-10/yr = 2.63E-6/yr, which is less than the RG 1.174 acceptance guideline of 1E-5/yr for Region II ("Small Change" in risk).

Table B-2

INTERNAL FLOOD CDF CONTRIBUTIONS [B-7]

Accident Class	Core Damage Frequency(1)	% of CDF
IA	5.4E-9	1%
ID	3.6E-8	8%
II	4.3E-7	91%
Total	4.7E-7	100%

Notes.

(1) All frequencies in events per reactor year.

Table B-3

ESTIMATE OF QUAD CITIES LERF DUE TO EXTERNAL EVENTS

External Event Accident Type	CDF	Conditional LERF Probability[1]	LERF
Loss of decay heat removal	5.39E-5[4]	3.26E-4	1.76E-8
Seismic-induced ATWS	3.60E-7[5]	4.12E-2	1.48E-8
Wide spread failure of seismic safe shutdown SSCs	4.80E-7[6]	1.00[2]	4.80E-7
Other scenarios (e.g., loss of coolant makeup, SBO, fire-induced ATWS, etc.)	1.37E-5[7]	1.35E-1[3]	1.85E-6
Totals	6.84E-5/yr	N/A	2.36E-6/yr

Notes:

(1) LERF conditional probabilities as a function of core damage accident type are taken from the Quad Cities Level 2 PRA (2002B) [B-6, Table 6.6-2]

(2) The LERF conditional probability for ISLOCA sequences (1.00) used to model LERF for seismic accidents involving wide-spread failure of safe shutdown SSCs. This is reasonable.

(3) The LERF conditional probability for Class IA accidents (loss of coolant makeup with RPV at high pressure) is used to model the other miscellaneous sequence types This is judged conservative.

(4) Sum of fire-induced and seismic induced loss of decay heat removal scenarios

(5) Seismic induced CDF (2.4E-6/yr) * 0.15

(6) Seismic induced CDF (2.4E-6/yr) * 0.20

(7) Seismic induced CDF (2.4E-6/yr) * 0.30 + fire-induced CDF (6.60E-5/yr) * (0.43 + 0.39 + .114)

Table B-4

ESTIMATE OF QUAD CITIES LERF DUE TO INTERNAL FLOODING EVENTS

Internal Flood Accident Class	CDF	Conditional LERF Probability[(1)]	LERF
Class IA	5.4E-9	1.35E-1	7.3E-10
Class ID	3.6E-8	0.0	0.0
Class II	4.3E-7	3.26E-4	1.4E-10
Totals	4.7E-7/yr	--	8.7E-10/yr

Notes:

(1) LERF conditional probabilities as a function of core damage accident type are taken from the Quad Cities Level 2 PRA (2002B) [B-6, Table 6.6-2]

REFERENCES

[B-1] Kennedy, R.P., "Overview of Methods for Seismic PSA and Margin Analysis Including Recent Innovations", Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan, August 1999. Available from OECD Nuclear Energy Agency, La Seine St.-Germain, 12 Boulevard des Iles, F-92130 Issy-les-Moulineaus, France

[B-2] ERIN Engineering and Structural Mechanics Consulting, Summary of Limerick Generating Station Seismic Margins Insights Evaluation, ERIN Report No. C0467010033-4801, June 2002.

[B-3] Electric Power Research Institute, Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue, NP-6395-D, April 1989.

[B-4] Quad Cities Fire IPEEE Insights and Sensitivities, ERIN Report no. R134-98-04.R08, June 1999.

[B-5] Energy Research, Inc., Technical Evaluation Report on the Review of the IPEEE at Quad Cities Nuclear Power Station, Units 1 and 2, Report No. ERIN/NRC 00-502, November 2000.

[B-6] Quad Cities Detailed Level 2 PRA Evaluation, QC-PSA-015, Volume 1, June 2002.

[B-7] Quad Cities Internal Flood Evaluation Summary and Notebook, QC-PSA-012, Rev. 1 (DRAFT), December 2002.