Exelon Generation®

4300 Winfield Road
Warrenville, IL 60555
630 657 2000 Office

RS-17-111

July 27, 2017

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

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

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

References:

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. NRC, "Request for License Amendment to Revise Technical Specifications Section 5.5.12 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies," dated April 27, 2017

2. Letter from K. J. Green (U.S. NRC) to B. C. Hanson (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 – Request for Additional Information Concerning Permanent Extension of Type A and Type C Leak Rate Test Frequencies (CAC. Nos. MF9675 and MF9676) (RS-17-051)," dated June 29, 2017

3. Letter from K. J. Green (U.S. NRC) to B. C. Hanson (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 – Request for Additional Information Concerning Permanent Extension of Type A and Type C Leak Rate Test Frequencies, Set No. 2 (CAC. Nos. MF9675 and MF9676) (RS-17-051)," dated July 17, 2017

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

The NRC requested additional information that is needed to complete review of the proposed change in References 2 and 3. In response to this request, EGC is providing the attached information. Specifically, the response to Reference 2 is provided in Attachment 1, and the response to Reference 3 is provided in Attachment 2.

EGC has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Attachment 1 of Reference 1. The additional information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

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

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

Respectfully,

David M. Gullott
Manager – Licensing

Attachments:
1. Response to Request for Additional Information dated June 29, 2017
2. Response to Request for Additional Information dated July 17, 2017

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

ATTACHMENT 1

Response to Request for Additional Information dated June 29, 2017

QC-LAR-04, "Quad Cities ILRT RAI Response"

RM DOCUMENTATION NO. QC-LAR-04	**REV:** 0	**PAGE NO.** 1	

STATION: Quad Cities Nuclear Power Station (QCNPS)

UNIT(S) AFFECTED: 1 and 2

TITLE: Quad Cities ILRT RAI Response

SUMMARY: Responses to NRC Request for Additional Information (RAI) for Quad Cities.

This is a Category I Risk Management Document in accordance with ER-AA-600-1012 Risk Management Documentation, which requires independent review and approval.

[ ] Review Required after Periodic Update

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

Electronic Calculation Data Files: N/A

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

This RM documentation supersedes: **N/A**

Prepared by:	John E. Steinmetz (Print)	/ John E. Steinmetz (Sign)	/ 7/26/17 (Date)
Prepared by:	Felipe Gonzalez (Print)	/ Felipe Gonzalez (Sign)	/ 7/26/17 (Date)
Reviewed by:	Grant Teagarden (Print)	/ Grant Teagarden (Sign)	/ 7/26/17 (Date)
Reviewed by: (Independent Review)	Don Vanover (Print)	/ Donald E. Vanover (Sign)	/ 7/26/17 (Date)
Approved by:	Jeff Stone (Print)	/ Jeff Stone (Sign)	/ 7/26/17 (Date)

NRC REQUEST FOR ADDITIONAL INFORMATION

EXELON GENERATION COMPANY, LLC

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 and 50-265

By letter dated April 27, 2017, Exelon Generation Company, LLC (EGC), submitted a license amendment request (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17121A449). The proposed amendment would modify Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow for the permanent extension of the Type A Integrated Leak Rate Testing (ILRT) and Type C Leak Rate Testing frequencies.

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. A draft request for additional (RAI) was transmitted by email to Mr. Mitch Mathews on June 16, 2017. A clarification call was held between members of your staff and NRC staff on June 22, 2017. As a result of the call, the staff revised RAIs 1b, 1e, and 3c to reduce ambiguity and more clearly state the NRC staff's request. Based on a discussion with Mr. Mathews, it was agreed that EGC will provide a response to the RAI within 30 days from the date of this letter.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the goal of efficient and effective use of staff resources.

NRC RAI 1a

1. EGC conducted a peer review of the internal events probabilistic risk assessment (PRA) in February 2017. EGC stated that the Facts and Observations (F&Os) from the 2017 peer review have not yet been resolved. With regard to the impact of the F&Os, provided in Table A-2 of Attachment 3 to the LAR, on the ILRT application address the following:

   a) F&O 1-9 identifies seven pre-initiator Human Failure Events (HFEs) potentially lacking documentation or detailed Human Reliability Analysis (HRA). Confirm that detailed HRA was performed for each of the listed pre-initiators, or alternatively, describe each pre-initiator and provide justification of its assumed failure probability estimate provided in the resolution to this F&O.

Response to 1a: The (Standard Requirements) SRs (HR-I1, HR-D2, HR-D5) associated with F&O 1-9 are MET to Capability Category I, which is considered adequate for this ILRT application. The seven identified pre-initiators are grouped with other similar actions due to the similarity of the actions, Quad Cities procedures, training, and test and maintenance practices. There are detailed calculations for each group of pre-initiators and all pre-initiators within a given group fall under the same calculated human error probability associated with that group. The disposition for each of the seven identified pre-initiators is listed below:

- 1HI-HPCI-TRA--HU - HPCI Misaligned
  This pre-initiator utilizes a detailed calculation applied from a similar action within the

same group. The HEP of 8.00E-04 is developed under Calculation A.2.13 1CSPMPTRA-----HU, *Preinit: Misalignment of CS Pump Train A.* This is listed in the Related Human Interactions section of Calculation A.2.13. This is documented in Appendix K, Section K.6 of the HRA notebook.

- BEPHU-1/2EDG-H-- - Failure To Restore EDG ½ to Operability After Maintenance
  This pre-initiator utilizes a detailed calculation applied from a similar action within the same group. The HEP of 8.00E-03 is developed under Calculation A.2.14 1EPHU-1-EDG-H--*, Preinit: Failure to Restore EDG 1 to Operability after Maintenance* and is grouped with BEPHU-1/2EDG-H--. This is documented in Appendix K, Section K.6 of the HRA notebook.

- BSSMV2-29011--H-- - Failure to Align SSMP to Operability After Maintenance
  This pre-initiator utilizes the same detailed calculations as Calculation A.2.13 1CSPMPTRA-----HU, *Preinit: Misalignment of CS Pump Train A*. It is listed in the calculation under the Related Human Interactions section of Calculation A.2.13. This is documented in Appendix K, Section K.6 of the HRA notebook.

The following pre-initiators also have detailed calculations for them. However, the four pre-initiators listed below were found to have incorrect values in the model as compared to their detailed calculations. The errors are explained below, but they are all minor and do not affect this application.

- 1CAHU263-52ABHCC - CAS Pressure Switches 52A and 52B Miscalibrated
  This pre-initiator has a probability of 8.00E-05 in the model. The HRA notebook documents the HEP at 4.95E-5. Correcting the model error by using the HEP from the HRA notebook results in a lower CDF. The impact to this application is considered negligible and conservative. When corrected, CDF decreases by 1% and LERF remains the same.

- 1CAHU263-23--HCC - CAS Level Transmitters 23A, B, C and D Miscalibrated
  This pre-initiator utilizes a detailed calculation applied from a similar action within the same group. This is documented in Appendix K, Section K.6 of the HRA notebook. However, the HEP is mistakenly included in the model with a value of 8.00E-05, it should have a value of 2.2E-04 as listed in the detailed Calculation A.2.7 1CAHU263-23ACH--, *Preinit: CAS Level Transmitters 23A and 23C Miscalibrated*. The corrected value of 2.20E-04 leads to an insignificant CDF increase of 0.35%, LERF remains the same.

- 1RSMV1001-4B-VHU - Closure of RHRSW to 1B RHR HX MO 1-1001-4B
  This pre-initiator utilizes the same detailed calculation as Calculation A.2.13 1CSPMPTRA-----HU, *Preinit: Misalignment of CS Pump Train A*. It is listed in the calculation under the Related Human Interactions section of Calculation A.2.13. This is documented in Appendix K, Section K.6 of the HRA notebook. However, it is mistakenly included in the model with a value of 1.00E-04 when the value should be 8.00E-04. The corrected value of 8.00E-04 leads to an insignificant CDF increase of 0.1%, while LERF remains the same.

- 1RSMV1001185BVHU - Closure of RHRSW From 1B RHR HX
  This pre-initiator utilizes the same detailed calculations as Calculation A.2.13 1CSPMPTRA-----HU, *Preinit: Misalignment of CS Pump Train A*. It is listed in the

calculation under the Related Human Interactions section of Calculation A.2.13. This is documented in Appendix K, Section K.6 of the HRA notebook. However, it is mistakenly included in the model with a value of 1.00E-04 when the value should be 8.00E-04. The corrected value of 8.00E-04 leads to an insignificant CDF increase of 0.1%, while LERF remains the same.

F&O 1-9 identified seven risk significant pre-initiators for which there appeared to be no detailed calculations. However, upon review, these seven pre-initiators are based on (and grouped with) other similar actions for which there are detailed calculations. There were also four errors found in the pre-initiator values implemented in the model. These errors have an insignificant decrease in CDF and no impact on LERF and therefore negligible impact on the ILRT application.

F&O 1-9 does not impact the conclusions of this application.

NRC RAI 1b

b) F&O 1-18 found that significant accident sequences were not reviewed to support equipment operation or operator actions during accident progression to reduce large early release frequency (LERF). EGC stated that it performed a review of the top 10 sequences, which constitute 90 percent of LERF. EGC dispositioned three of the sequences by stating that they have no impact on the application because they don't impact Class 3b frequency, being assigned to EPRI Class 7 or 8. EGC further stated that the "remaining LERF sequences would also have an insignificant impact on this application as the ILRT methodology is sensitive to changes in [core damage frequency] CDF, not LERF." The NRC staff notes that the internal events CDF to LERF ratio is used in the application to estimate the LERF from external hazards, including seismic and fire, and therefore, the internal events LERF has more than a minimal impact on the estimated total LERF. Address the impact of this F&O on total LERF.

Response to 1b: It is recognized that the LERF to CDF ratio is used for this application, however the SR's associated with this finding were determined to meet Capability Category I which is sufficient for this application. In Section 3.5.2 of the QC Summary Notebook (QC-PSA-013), the LERF Sequences which contribute 95% of LERF are reviewed. Section 3.5.1 of the same notebook reviews the Top 10 LERF cutsets, which happen to contribute about 90% to LERF. Addressing F&O 1-18 more fully would reduce the FPIE LERF, LERF to CDF ratio, and estimated total LERF.

For the above reasons, F&O 1-18 does not adversely impact the conclusions of this application.

NRC RAI 1c

c) F&O 2-10 found that supporting requirement QU-B3 was not met because the truncation limit of 1E-12 used in the LERF quantification did not show that the model results converged. Based on a sensitivity study on truncation levels cited by EGC in response to this F&O, the LERF would converge to 2.19E-7/year, instead of the LERF of 1.97E-7/year that was provided in the LAR (an increase of 11 percent). EGC stated that this F&O would contribute only to a small change to total LERF only. The NRC staff notes that the internal events CDF to LERF ratio is used in the application to estimate the LERF from external hazards, including seismic and fire, and therefore the internal

events LERF has more than a minimal impact on the estimated total LERF. Provide a quantitative estimate of impact on total LERF from this F&O.

Response to 1c: Per Table A-2 of Attachment 3 to the LAR, LERF was demonstrated to meet the 5% convergence criteria for a LERF value of 2.10E-7/yr (truncation at 2E-13/yr). Appropriate convergence at 2.10E-7/yr was demonstrated based on the 2.19E-7/yr quantification at a truncation of 2E-14/yr. Therefore, the LERF increase associated with quantification at a lower truncation level demonstrating convergence is 6.6% (i.e., (2.10E-7/yr – 1.97E-7) / 1.97E-7/yr). An estimate of the quantitative impact of using a FPIE LERF value of 2.10E-07/yr is included in the responses to RAI 6.

NRC RAI 1d

d) F&O 3-2 identified deficiencies regarding consideration of common cause failures (CCF) or maintenance activities for special initiators, which typically involve support system failures. The peer review stated that potential common cause and alignment are generally addressed in the fault tree supporting the initiating events, however the effects of maintenance are not. The peer review provided few examples: maintenance events were not included for the service water and the component cooling water (CCW) initiating events, and CCF of heat exchangers and pumps was missing for CCW initiators. EGC stated that special initiator systems would also include circulating water and instrument air systems. For each risk-significant special initiator, confirm that maintenance activities and CCF are addressed, and reflect the as-operated plant, and if not, assess, preferably quantitatively, the impact on CDF and LERF.

Response to 1d:

The LAR submittal also showed that the SRs associated with F&O 3-2, except for the documentation SR, were met at Capability Category II. For this application, Capability Category I is adequate.

i. Reg. Guide 1.200 Rev. 2 Table A-2 documents an Issue "Initiating events from common cause or from both routine and non-routine system alignments should be considered. The staff position was that for Cat II: "When performing the systematic evaluation required in IE-A5, INCLUDE initiating events if the equipment failures result from a common cause, or from routine system alignments resulting from preventive and corrective maintenance.

Loss of Instrument Air and Circulating Water Systems were considered for special initiator fault trees, but special initiator fault trees were not included in the FPIE PRA model. The Loss of Instrument Air (%TIA) initiating event is represented by a point estimate of 1.50E-02/yr. This point estimate is based on the loss of instrument air initiator frequency calculated using the NUREG/CR-5750 Loss of Instrument Air (BWR) frequency distribution as a prior and performing a Bayesian update with the Quad Cities experience. The Loss of Circulating Water initiator is subsumed in the Loss of Condenser Vacuum (%TC) initiating event. The Loss of Condenser Vacuum initiating event is represented by a point estimate of 7.76E-02/yr based on NUREG/CR-6928 data and a Bayesian update with the Quad Cities experience.

There are only three special initiator fault trees, RBCCW, TBCCW and TSW, in the FPIE PRA model. The Fussell-Vesely value for each special initiator is shown below:

- Loss of Reactor Building Closed Cooling Water (RBCCW) FV = 5.56E-05
- Loss of Turbine Building Closed Cooling Water (TBCCW) FV = 1.48E-02
- Loss of Service Water (TSW) FV = 7.93E-02

Risk Significant Initiating Events may be defined as basic events having FV greater than 5E-03[1]. Thus, the Loss of TSW and TBCCW Initiating Events are defined as risk significant and will be the focus of this response.

With respect to maintenance terms, the TSW and TBCCW special initiator fault trees do have maintenance terms in the model logic and were apparently missed by the peer review team. The following maintenance terms are included:

- Loss of TBCCW: Maintenance unavailability for individual pumps and heat exchangers
    - 1TBPM1-3801A-M-- (TBCCW PUMP 1-3801A UNAVAILABLE DUE TO MAINTENANCE)
    - 1TBPM1-3801B-M-- (TBCCW PUMP 1-3801B UNAVAILABLE DUE TO MAINTENANCE)
    - 1TBHE1-3802A-M-- (HEAT EXCHANGER 1-3802A UNAVAILABLE DUE TO MAINTENANCE)
    - 1TBHE1-3802B-M-- (HEAT EXCHANGER 1-3802B UNAVAILABLE DUE TO MAINTENANCE)
- Loss of TSW: Maintenance unavailability for individual pumps
    - BSWPM1-3901A-M-- (SW PUMP 1-3901A UNAVAILABLE DUE TO MAINTENANCE)
    - BSWPM1-3901B-M-- (SW PUMP 1-3901B UNAVAILABLE DUE TO MAINTENANCE)
    - BSWPM1/2-3901M-- (SW PUMP 1/2-3901 UNAVAILABLE DUE TO MAINTENANCE)
    - BSWPM2-3901A-M-- (SW PUMP 2-3902A UNAVAILABLE DUE TO MAINTENANCE)
    - BSWPM2-3901B-M-- (SW PUMP 2-3902B UNAVAILABLE DUE TO MAINTENANCE)

With respect to CCF modeling, common cause basic events in the TSW Initiating Event model logic include:

- Failure to run of two or more normally running pumps. Basic Event BSWPM-3901---XCC probability of 1.11E-03. This event uses the Beta Approach to estimate the CCF for 2 or more SW pumps fail to run (FTR) for a year.
- Failure to start of two or more standby pumps. Basic Event BSWPM-3901---ACC probability of 3.72E-05. This event uses the Beta Approach to estimate the CCF for 2 or more SW pumps fail to start (FTS).

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(1) D. True et al., “PSA Applications Guide,” Electric Power Research Institute, TR-105396, August 1995.

- Plugging of two or more SW strainers. Basic Event BSWFL2STRNSIEPCC probability of 2.76E-04. This event uses the Beta Approach to estimate the CCF for 2 or more SW strainers plugging over the course of a year.

Common cause is considered in the Loss of Service Water initiating event fault tree. Therefore, this initiating event fault tree is considered applicable and would meet Capability Category II.

Common cause basic events in the TBCCW Initiating Event model logic include:

- TBCCW fails due to leakage. Basic Event 1TBSYSTEMIE--L-- with failure probability of 3.40E-4 and failure to recover of 0.5. This scenario has a yearly initiating event contribution of 1.70E-4/yr.
- TBCCW SW TCV closed due to control signal fault. Basic event 1SWAV13903-IEV-- with a failure probability of 1.31E-03 (no recovery credited). This scenario has a yearly initiating event contribution of 1.31E-3/yr.

As noted by the peer review team, CCF of the TBCCW running and standby pumps and the CCW heat exchangers are not modeled. The potential impact of modeling these CCF terms is assessed below.

<u>TBCCW Quantitative Sensitivity Case</u>

For a sensitivity case, the NRC Quad Cities SPAR Model TBCCW pump failure to run CCF was applied to the EGC FPIE Quad Cities TBCCW Fault Tree. A failure to start common cause event is not applicable as one pump is normally running and one in standby. The NRC SPAR model does not have this CCF event. The SPAR Model also includes heat exchanger common cause failure events as shown in the table below. Both the TBCCW pump failure common cause failure to run and the heat exchanger basic events were added to the EGC FPIE model. The SPAR model IE CCF basic events and failure probabilities are from page B-242 of Standardized Plant Analysis Risk Model for Quad Cities 1&2, Revision 8.50 dated March 2017.

SPAR MODEL BASIC EVENT	DESCRIPTION	PROBABILITY
IE-TBC-MDP-CF-FR	TBCCW MDPS FAIL FROM COMMON CAUSE TO RUN – IE	5.2E-4
IE-TBC-HTX-CF-HTXS	TBCCW HEAT EXCHANGERS FAIL FROM COMMON CAUSE - IE RANDOM	1.1E-4
IE-TBC-HTX-CF-HTXSE	TBCCW HEAT EXCHANGERS FAIL FROM COMMON CAUSE (ENVIRONMENTAL)	3.1E-5

The Quad Cities FPIE base CDF model TBCCW frequency initiating event fault tree is 2.33E-03/yr. Following addition of the SPAR Model CCF basic events the TBCCW initiating event frequency is 2.89E-03/yr. The frequencies are found by quantifying the TBCCW initiating event fault tree at a truncation limit of 1E-12/yr.

Results are shown below.

	BASELINE (/YR.)	WITH CCF EVENTS (/YR.)	INCREASE (/YR.)	INCREASE (%)
CDF	2.91388E-06	2.91838E-06	4.5E-09	0.15
LERF	1.97101E-07	1.97228E-07	1.3E-10	0.06

The estimated CDF and LERF increases associated with including additional CCF events are small. If assessed in the ILRT risk assessment, the CDF and LERF increase would apply to the ILRT base case frequency (i.e., 3-per-10 year) as well as the extended interval (i.e., 1-per-15 year) and would have a negligible change to the ILRT quantitative results.

NRC RAI 1e

e) F&O 3-9 found that the PRA data analysis spanned only 4 years of plant-specific experience and it did not justify exclusion of plants events that occurred prior to January 2010. This F&O appears to apply to initiating event frequencies, as well as to equipment failures probabilities and unavailabilities. EGC discussed qualitatively the impact on the application of this F&O for two of the initiators, general transients and loss of offsite power. However, no discussion of impact on the application from not including plant operational experience for other initiating events or for equipment failure probabilities and unavailabilities was provided in the LAR. Assess and justify the impact of this F&O on the estimated CDF and LERF from any plant-specific operational experience related to other initiating event frequencies that were not discussed in the LAR, and from plant specific operational experience related to equipment failure probabilities and unavailabilities.

Response to 1e:

Regarding initiating events, the reason only two of the initiators were discussed is because in the period from Jan. 1, 2005 to Dec. 31, 2016 only those two types of initiators occurred.

Regarding equipment failure probabilities, it is recognized that a significant amount of traceable, representative, plant-specific failure data is preferred. EGC considers that three years of failure data is the preferred minimum to accurately represent future plant operation. Quad Cities used the past four years of plant specific failure probability data. Plant-specific failure data prior to the four year data period was captured in data sources such as NUREG/CR-6928 which are used to calculate generic prior failure probabilities. Since the prior Quad Cities data was captured in these generic sources used in the data update process, exclusion of this older Quad Cities data from “new” data is judged acceptable.

Regarding maintenance data, maintenance practices have the potential to change with more regularity than some other types of data (e.g., failure data), therefore a shorter data collection period is generally appropriate. The last 3 to 5 years are judged to be reflective of future plant operational practices. Therefore, Quad Cities used a 4 year data collection period.

NRC RAI 2

2. Section 4.2 of Attachment 3 of the LAR indicates that there are no "substantive differences" between Unit 1 and Unit 2 that are judged to affect the conclusions of the PRA, and that the Unit 1 PRA results are judged representative for Unit 2. Provide a brief description of the differences between the units, particularly those differences that might impact internal events and internal flooding risk, and provide justification for concluding that Unit 1 PRA results are representative of Unit 2.

Response to RAI 2:

A brief description of the differences between Unit 1 and 2 is provided. Additional information with respect to differences that impact fire risk at the site is also provided since that is the subject of the RAI 3b response.

Unit Differences that Impact Internal Events and Internal Flood Risk

There are no significant differences relevant to the FPIE model between Unit 1 and Unit 2 at Quad Cities. The Unit 2 model contains some power supply differences and in the internal flooding model there are minor asymmetries in potential spray targets in certain rooms, but when the Unit 2 model is quantified with these differences, the CDF and LERF results are the same as for Unit 1.

Unit Differences that Impact Fire Risk

Quad Cities Nuclear Power Station Unit 1 and Unit 2 are generally arranged the same, which includes equipment and cable routing. The exception is that the Main Control Room and Aux. Electric Equipment Room are shared between the units and are located in the Service Building which is located south of Unit 1. Unit 2 is located north of Unit 1. Therefore, Unit 2 cables are routed across Unit 1 to terminate in the Service Building. This difference is largely offset, because the Unit 2 cables are mostly routed to the Service Building via the Unit 2 Cable Tunnel. However, given the Unit 2 cables are routed across Unit 1 to the Service Building, there are a larger number of plant locations in the Fire PRA that may potentially impact Unit 2 cables.

Additionally, there is a difference with the QC Safe Shutdown Procedures (QCARPs) for shutdown of Unit 1 and 2 in the case of a Service Building fire. QCARP 0050-01 directs using the Safe Shutdown Makeup Pump (SSMP) to safely shutdown Unit 1, while QCARP 0050-02 directs using RCIC to safely shutdown Unit 2.

The IPEEE Fire PRA Insights and Sensitivities Report 134-98-04.R08 discusses some of the key differences between Unit 1 and 2. Some excerpts are reproduced here:

"For Unit 2, the base CDF is 7.13E-05 per year. Similar to QC1, the dominant contributor for QC2 involves a large RFP fire (FZ086B), and the ten largest scenarios comprise two-thirds of the overall CDF."

"The large oil fires involving the Reactor Feedwater Pumps are the dominant risk contributors for both units because of the location of the cables and circuits associated with the RHRSW system."

"The original Fire IPEEE submittal reported a lower CDF contribution for Unit 2 as compared to Unit 1. This is in contrast to the upgraded fire analysis results which indicate the opposite. In addition, a review of the dominant risk contributors presented in Tables 2-1a and 2-2 shows two notable asymmetries in the risk profile. The risk contribution from RFP fires in Unit 2 is approximately 10% higher than in Unit 1. This is because of the specific cable routing of the power supply circuit to MCC 29-2 which is challenged by postulated Unit 2 RFP fires. The equivalent MCC in Unit 1 (MCC 19-2) is not exposed to such a challenge. The Unit 2 results also show a 4% risk contribution from an air compressor because of the proximity of cable trays containing critical circuits for HPCI, SSMP, and one train of CS and RHR. Such an exposure does not exist in the Unit 1 analysis."

"The base scenarios for Unit 1 were also examined to determine the dominant fire zones and Appendix R fire areas. In the first case, twelve fire zones comprised 95% of the overall CDF. The largest contributor to the Unit 1 CDF is the Unit 1 Turbine Building Ground Floor (South), or FZ082, which encompasses the reactor feed pump areas. For Unit 2, the RFP area (Unit 2 Turbine Building Ground Floor - North), or FZ086, is also the largest contributor."

"The second largest contributor to the Unit 1 CDF is the Main Control Room, FZ054. A severe fire requiring evaluation *[sic; should state "evacuation"]* of the control room comprises the majority of this contribution. The dominant control room scenario that did not require abandonment involves a postulated fire in panel 901-3/902-3. Such a fire challenges the functionality of ADS, one train of RHR, and the torus/drywell vent valves. For Unit 2, the Main Control Room is the third largest contributor, while the Unit 2 Cable Tunnel (FZ081) is the second largest contributor."

Differences between the Unit 1 and Unit 2 Fire CDF from the IPEEE are further addressed in RAI Response 3b.

NRC RAI 3a

3. Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2-A (ADAMS Accession No. ML14024A045) states that "[w]here possible, the analysis should include a quantitative assessment of the contribution of external events (for example, fire and seismic) in the risk impact assessment for extended ILRT intervals. For example, where a licensee possesses a quantitative fire analysis and that analysis is of sufficient quality and detail to assess the impact, the methods used to obtain the impact from internal events should be applied for the external event." EPRI TR-1009325, Revision 2-A further states that the "assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval."

   In Section 5.7 of Attachment 3 to the LAR, EGC performed an assessment of external event contribution. To assess the fire risk, EGC stated that "a quantifiable fire PRA model meeting an appropriate level of ASME/ANS Standard is under development for QCNPS." It further stated that "'[t]he QCNSP Fire PRA updated in 1999 as part of the revised QCNPS Individual plant Examination of External Events (IPEEE) Submittal is judged to be adequate to support the ILRT External Events quantitative risk assessment."

   a) If an estimate of fire CDF and LERF is available for the current fire PRA under development, provide those values with a summary description of the status of the fire PRA under development.

Response to RAI 3a:

An estimate of Fire CDF and LERF for the current (2017) Fire PRA under development is not available. Quad Cities Power Station does not have a Fire PRA that includes all of the NUREG/CR-6850 tasks, and the Fire PRA does not include the current industry guidance. An interim Fire PRA was developed in 2010 based on the IPEEE and the internal events model at that time. The interim Fire PRA results have not been analyzed in detail to meet the requirements of the ASME/ANS Standard for Fire PRAs, and the postulated fires have not been modeled in sufficient detail to accurately quantify the Fire CDF and LERF. The Fire PRA has been used for qualitative assessments and quantitative insights for delta risk calculations. Given the interim status of the 2010 Fire PRA, the IPEEE has been used to estimate the Fire PRA CDF and LERF for the ILRT risk assessment which only needs a reasonable first order approximation of total Fire CDF. The interim Fire PRA was updated and peer reviewed in 2013 against Part 4 of the ASME/ANS RA-Sa-2009 Standard. Because of the rapidly changing state-of–knowledge of Fire PRAs at that time, the purpose of the peer review was to have the methods being applied assessed and to glean insights from the peer review team regarding the use of industry methods that were in draft status. The peer review resulted in a number of F&Os. The Fire PRA was scheduled to be completed in 2016 to resolve the peer review F&Os and to incorporate the latest industry guidance (e.g., NUREG-2178 heat release rates, NUREG-2169 fire ignition frequencies, NUREG/CR-7150 circuit failure probabilities and durations). However, the Fire PRA update was discontinued upon EGC's announcement to close the Quad Cities Power Station.

Following EGC's decision to keep the statin in operation, the current status is that the Fire PRA update is now in progress, with the work just being re-initiated last month (June 2017). There are still F&Os to be resolved, more detailed fire modeling is needed to get accurate results, and the latest industry guidance has not been incorporated; therefore, an estimate of Fire CDF and LERF for the current fire PRA that is representative of the Quad Cities Power Station is not available. The IPEEE fire results are judged to provide the most reasonable order of magnitude estimate for the contribution associated with fire risk.

NRC RAI 3b

b) In Section 5.7.1 of Attachment 3, EGC uses the Unit 1 fire CDF results to estimate the risk from the external hazards for the ILRT extension application for both Unit 1 and Unit 2, but no justification was provided. As cited in the original IPEEE (Section 4.2 of "Technical Evaluation Report on the Review of the Individual Plant Examination of External Events at Quad Cities Nuclear Power Station, Units 1 and 2," dated January, 2001, ADAMS Accession No. ML011410547), the Unit 2 fire CDF was 7.13E-5/year, which is higher than the Unit 1 fire CDF of 6.6E-5/year. Justify the applicability of the Unit 1 fire CDF to estimate Unit 2 risk or provide an updated estimate for Unit 2.

Response to RAI 3b:

The Unit 2 Fire CDF is the most limiting between the two units, with the Unit 2 Fire CDF approximately 8% higher. Use of the Unit 2 results would have a small quantitative impact (increase) on the ILRT risk results. This quantitative impact using the Unit 2 Fire CDF of 7.13E-5/year is developed as follows, based on updating text from Section 5.7.1 of Attachment 3 of the LAR:

Revised (Updated) Section 5.7.1

The QCNPS **Unit 2** CDF contribution due to internal fires in the unscreened fire areas was calculated at **7.13E-5/yr.** The breakdown of the QCNPS fire risk profile for **Unit 2 is assumed to be similar to Unit 1**, as follows:

TABLE 5.7-1

QCNPS UNIT 1 & 2 FIRE RISK PROFILE

RELEASE TYPE	U1 CONTRIBUTION	U1 CDF	U2 CDF[1]
Fire-induced loss of decay heat removal scenarios	80.4%	5.31E-05	5.73E-05
Fire-induced loss of inventory control scenarios (RPV at low pressure)	4.3%	2.84E-06	3.07E-06
Fire-induced loss of inventory control scenarios (RPV at high pressure)	3.9%	2.57E-06	2.78E-06
Other fire-induced scenarios	11.4%	7.52E-06	8.13E-06
Totals	**100%**	**6.60E-05**	**7.13E-5**

Note to Table 5.7-1:

(1) U2 CDF calculated based on total U2 CDF and U1 Contribution.

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

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

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

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

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

TABLE 5.7-2

QC FPIE LERF CONTRIBUTION FROM DECAY HEAT REMOVAL SEQUENCES

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

Note to Table 5.7-2:

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

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

As outlined in Table 5.7-1, fire-induced loss of decay heat removal scenarios is **5.73E-5/yr**. Assuming that 95% of the fire-induced loss of decay heat removal scenarios have the GE declared 'early", the fire induced release frequency that cannot be LERF is:

5.73E-5/yr. * 0.95 = **5.44E-5/y**r.

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

Total CDF – Decay Heat Removal CDF = **7.13E-05/yr. – 5.44E-5/yr**
Fire CDF applicable to 3b frequency = **1.69E-05/yr**.

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

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

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

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

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

Since ILRT delta risk is primarily a function of CDF, it could be assumed that the total impact from the ILRT fire risk contribution is bounded by assuming a fire multiplier factor of **8.7** (**1.69E-05/yr.** ÷ 1.94E-06/yr.) compared to the internal events evaluation alone. However, the fire multiplier will be applied towards the full FPIE CDF (2.92E-06/yr) which includes Class II non-LERF sequences. To better represent the impact of fire risk for the ILRT assessment where non-LERF frequencies can be excluded, the fire multiplier of **8.7** can be discounted by the ratio of FPIE CDF with Class II sequences removed to the full FPIE CDF (i.e., 1.94E-06/yr ÷ 2.92E-06/yr. = 0.665) to give a fire multiplier of **5.8**.

This new fire multiplier is further propagated through the ILRT risk calculations in the response to RAI 6.

NRC RAI 3c

c) In the LAR, EGC provided an estimate of 6.6E-5/year for the fire CDF from the IPEEE. To estimate the change in LERF due to the ILRT, the calculation of the Class 3b frequency used a reduced frequency of 1.56E-5/year by eliminating 95% of the fire-induced loss of decay heat removal scenarios, based on an assumption that these scenarios would have the general emergency declared "early," such that the release

would be considered non-LERF. The NRG staff notes that some of these non-LERF scenarios could become LERF if the containment had an undetected leak, and that EGC's assumption seems to be inconsistent with the guidance in EPRI TR 1009325 Revision 2-A, which allows removing only "sequences that either may already (independently) cause a LERF or could never cause a LERF." Additionally, this discussion on fire-induced loss of DHR scenarios appears to be based on internal events LERF scenarios, and may not be directly applicable in the context of fire scenarios. Justify the reduction in fire CDF in the calculation of the Class 3b frequency and explain how it is consistent with the guidance in EPRI TR 1009325 Revision 2-A, or remove this credit and provide an updated estimate of risk from external hazards.

Response to RAI 3c:

The internal events basis for the “early” declaration of a general emergency for loss of decay heat removal accident sequences is documented in Appendix G of the Quad Cities Level 2 PRA Notebook (QC PRA-015). This assessment is based on review of the Quad Cities Emergency Action Levels (EALs), and plant personnel interviews. It was concluded that the probability of failure to declare a general emergency at least 3 hours prior to containment failure is 5%. For such sequences, MAAP case results for the PRA show that containment failure will not occur until approximately 30 hours after the initiating event. Due to containment failure, the RPV injection is assumed to be lost, leading to core damage after approximately another 5 hours (t=35 hours). Following core damage, it would take additional time (on the order of hours) for a release to exceed the threshold for large. Note that the time to complete evacuation of the emergency planning region is less than 5 hours. For a general emergency declaration by 3 hours prior to containment failure, evacuation will be complete hours before the on-set of core damage. Therefore, consistent with the EPRI guidance, any associated releases will not be LERF based on timing, and given the slow evolving nature of this scenario, a 5% LERF likelihood is judged to be conservative.

Fire-induced loss of DHR scenarios are similar to internal events loss of DHR scenarios where the slow developing nature of such events would result in an “early” declaration of a general emergency. Operating experience evidence shows that fires at plants are extinguished within an hour of initiation. After the extinguishment of a fire that induces a loss of decay heat removal scenario, there are still many hours of margin for the operating crew (as well as the manned TSC) to determine the need to declare a general emergency. Loss of DHR scenarios induced by internal events or fires would not deviate severely enough from each other to have significantly different general emergency declaration timings. In consideration of larger, uncontained fires, it is potentially more likely that a general emergency would be declared earlier given the expected increase in loss of unrecoverable systems/functions.

Typically operator action adjustments for fire scenarios must be considered in comparison to the internal event assessments. However, for operator actions credited for loss of DHR scenarios, the adjustments are low to negligible:

- Loss of DHR scenarios provide significant timing margin to assess and execute required actions
- Once the fire has been extinguished within the first hour, accessibility concerns for operator actions are minimized

It is also noted that the fire risk estimate used in the ILRT submittal from the IPEEE fire risk study would consider the potential impact of fires upon operator actions.

NRC RAI 4

4. Section 5.7 of Attachment 3 of the LAR states that high winds, tornadoes, external floods, transportation accidents, nearby facility accidents and other external hazards were not considered because of their negligible contribution to overall plant risk. This conclusion was reached based on the QCNPS Individual Plant Examination for External Events (IPEEE) analysis performed in 1999, based on the fact that the plant met the applicable Standard Review Plan requirement.

   a) Given that the external hazard information has not been updated since the IPEEE studies, discuss the applicability of the IPEEE conclusions to the current plant and its environs, considering each of the external hazards screened from this application and taking into account any updated risk studies and insights.

   b) In light of recent external flooding re-evaluation performed in response to Near-Term Task Force recommendations, provide technical justification for why the risk from external flooding is negligible, or provide, with justification, a conservative or bounding estimate of the impact of external flooding risk for the current application.

Response to RAI 4:

The IPEEE analysis of High Winds/Tornadoes, External Floods and Transportation and Nearby Facility Accidents are re-examined below taking into account updated risk studies and insights.

The latest external flooding re-evaluation performed in response to Near-Term Task Force (NTTF) recommendations is summarized in several documents. These include:

EGC letter RS-13-047 to the US NRC dated March 12, 2013. The subject of RS-13-047 is "Response to March 12, 2012, Request for Information Enclosure 2, Recommendation 2.1, Flooding, Required Response 2, Flooding Hazard Reevaluation Report." EGC letter RS-13-047 is the primary source of information for the External Floods response found below.

EGC letter RS-15-001 to the US NRC dated January 13, 2015. The subject of RS-15-001 is "Response to Request for Additional Information Regardiing Fukushima Lessons Learned – Flood Hazard Reevaluation Report."

High Winds and Tornadoes (including Tornado Missiles)

The major concern in a high-wind or tornado scenario are the wind loads imposed on the buildings/major structures and the potential for wind-generated missiles to disable systems or components necessary to shut down the plant of maintain the plant in a safe shutdown condition. There have been no major changes to the buildings/major structures or location of important to safety equipment within them since the IPEEE submittal in 1999. The only significant change is the addition of FLEX/HCVS equipment and procedures which provide the station with additional response capability to an event.

External Floods

Flood causing mechanisms were re-evaluated to address NTTF recommendations (Ref. EGC letter RS-13-047). Storm Surge, Seiche and Tsunami flood mechanisms are not applicable to QCNPS.

The following flood causing mechanisms were found bounded by the current design basis: Flooding in Streams and Rivers, Dam Breaches and Failures, Ice Induced Flooding and Channel Mitigation or Diversion.

UFSAR Section 3.4 describes actions to be taken in the event of a maximum probable flood.
As discussed in the Quad Cities UFSAR Section 3.4, "In the highly unlikely event that a maximum probable flood is predicted, steps to shutdown and cool the plant will be initiated a minimum of three days prior to the predicted time at which the water will go above plant grade elevation of 594.5 feet. This will reduce the decay heat from the reactor to a level which can be removed by natural circulation cooling between the reactor and the reactor cavities and storage pools. Once the reactor is shutdown and cooled down, the drywell and reactor vessel heads will be removed. The hatches in the bellows seal structure will then be closed and the reactor cavities and dryer-separator pools will be filled to the level of the spent fuel pool. Finally, the spent fuel pool gates will be removed permitting free circulation of water through the storage pools, reactor cavities and dryer-separator pools." Since the primary containment is opened (drywell head removed) in response to a bounding external flood, containment leakage rate has no impact on the event.

Dam Failures (bounded by current design basis)

A seismically induced upstream dam failure re-evaluation found a peak water surface elevation of 592.5 feet which is below the grade elevation of 595.0 feet. The UFSAR provides an assessment of down-stream dam failure. It was predicted that 90 hours are required to draw the river stage to elevation 565 ft from 572 ft at QCNPS following a catastrophic loss of lock miter gates. The ultimate heat sink captured volume is approximately 3 million gallons at an elevation of 574'9". Following a dam failure, approximately 2 days would be available to position portable pumps to make-up to the ultimate heat sink (UFSAR Section 9.2.5.2).

A Dam breach concurrent with a PMF is not bounded and is discussed under Flood Causing Mechanisms Not Bounded section below.

Ice Jam (bounded by current design basis)

The maximum water surface elevation reevaluation at QCNPS found an upstream ice breaching water level to be 573.7 ft, 21.3 ft below site grade. The maximum water surface elevation resulting from backwater from a downstream ice jam is 579.8 ft, 15.2 ft below site grade. Significant margin exists between calculated water surface elevations and site grade at QCNPS.

Channel Mitigation or Diversion

As indicated in the UFSAR, the authority to control the river is vested in the USACE. Should the need arise, Exelon will make the required notification to the USACE. There arrangements have been made and are detailed in the QCNPS emergency procedures.

Flood Causing Mechanisms Not Bounded (by the current design basis)

Local Intense Precipitation (LIP)

QCNPS performed detailed studies, designed and installed modifications to establish a zone of protection for all critical plant equipment from the LIP event. All exterior walls, penetrations and doors were examined for the effects of the LIP event. Structural modifications and multiple new commercial flood barriers have been installed to protect the critical plant equipment. These changes will prevent critical plant equipment from being adversely impacted by flood water intrusion as a result of a LIP event. Additionally, roof parapets have been modified on the Reactor Building, Turbine Building, and other critical roof structures to prevent excessive ponding thus reducing the structural load to an acceptable level.

Combined Effects (PMF + Dam Failure + Wind-Generated Waves)

The Combined Effects River Flood Scenario results in a maximum flood elevation of 605.0 ft (Maximum still water elevation = 600.9 ft + 4.1 ft of wind wave runup) which exceeds the probable maximum flood elevation described in Quad Cities UFSAR of 603 ft. However, the QCNPS response to the Mississippi River Flood has been reviewed in detail and the response steps outlined in Quad Cities UFSAR Section 3.4 are applicable for the Combined Effects River Flood Scenario. As discussed above, since the primary containment is opened (drywell head removed) in response to a bounding external flood, containment leakage rate has no impact on the event.

Hydrodynamic and Debris Loads

Hydrodynamic and Debris Loads have been evaluated and determined not to impact the ability of QCNPS to implement response procedures necessary to safely shut down the plant in accordance with the Quad Cities USFAR Section 3.4.

Transportation and Nearby Facility Accidents

The major concern for a transportation or nearby facility accident is the release of toxic material or a pressure load on station structures from a resulting explosion. As discussed in the UFSAR 6.4, the Quad Cities Control Room is protected from toxic hazards in accordance with Regulatory Guide 1.78. Facilities within five miles are periodically surveyed (the last survey performed in 2014) and the bounding hazards discussed in the UFSAR remain unchanged since 1999. While the Quad Cities UFSAR is focused on the toxic effects of this external event, the potential explosion effects were addressed in the IPEEE submittal. The station has made no changes to the major safety-related structures on-site, such as the Reactor Building or the Control Room envelope, or to the proximity of nearby highways or railways.

NRC RAI 5

5. Section 4.2.6 of EPRI TR-1009325, Revision 2-A states that, “[p]lants that rely on containment overpressure for net positive suction head (NPSH) for emergency core cooling system (ECCS) injection for certain accident sequences may experience an increase in CDF,” therefore, requiring a risk assessment.

a) Section 5.2.4 of EPRI TR-1009325, Revision 2-A includes guidance on performing this risk assessment and provides the following examples of accident scenarios to be considered:

- LOCA [loss-of-coolant accident] scenarios where the initial containment pressurization helps to satisfy the NPSH requirements for early injection in BWRs or PWR [pressurized-water reactor] sump recirculation
- Total loss of containment heat removal scenarios where gradual containment pressurization helps to satisfy the NPSH requirements for long term use of an injection system from a source inside of containment.

The NRC staff notes that the EPRI guidance cited above refers to all LOCA scenarios, not limited to LOCA initiators only. LOCA initiators are a relatively small contributor to the internal events CDF. However, other internal events initiators (e.g., transients) or conditions (e.g., station blackout) could result in a consequential LOCA.

Explain and justify how all the initiating events or conditions which correspond to the two EPRI guidance general event categories were considered in the ΔCDF estimate of 7.2E-8/year provided in the LAR for the loss of containment overpressure.

b) Describe and justify the PRA modeling and methodology used in the LAR to estimate ΔCDF due to loss of containment overpressure. The LAR states that the containment isolation failure was increased by the frequency of Class 3b in order to derive ΔCDF. Containment isolation failure is typically linked to LERF, so it is not immediately clear to the NRC staff how increasing containment isolation failure probability would produce a change in CDF.

c) If a new estimate of ΔCDF results from items a) and b) above, estimate the contribution to ΔLERF resulting from loss of containment overpressure impact on the ECCS pump performance, including the contribution from loss of containment overpressure to the risk from external hazards. Describe and justify the method used for estimating ΔLERF.

Response RAI 5:

The response to RAIs 5a, 5b, and 5c are interrelated and are therefore presented together. Based on further review of Quad Cities system capabilities, a revised estimate for ΔCDF is presented based on modeling refinements. A bounding approach to ΔLERF from the revised ΔCDF due to NPSH is included part of response to RAI 6.

The Quad Cities PRA model addresses loss of NPSH to Core Spray and RHR pumps for all applicable initiating events. Loss of NPSH from loss of containment heat removal Suppression Pool Cooling (SPC) and Shutdown Cooling (SDC) are considered for each initiating event and modeling is documented in the Quad Cities Success Criteria and Event Tree Notebooks. The Loss of NPSH modeling is developed for isolation failures, including a pre-existing leak, as well as spurious drywell spray actuation. The PRA model and documentation have been peer reviewed. In addition to the Quad Cities PRA, this approach has been applied in the Dresden PRA. The Dresden PRA has been peer reviewed. No findings or observations regarding loss of NPSH treatment were identified.

The RAI 5a request is to explain and justify how all the initiating events or conditions which correspond to the two EPRI guidance general event categories (i.e., LOCA, loss of containment heat removal) were considered in the ΔCDF estimate of 7.2E 8/year provided in the LAR for the loss of containment overpressure.

The information below addresses all the initiating events and conditions which correspond to the two EPRI guidance general event categories (i.e., LOCA and Loss of Containment Heat Removal scenarios). It should be noted that the QC FPIE PRA does not credit gradual containment pressurization to satisfy the NPSH requirements for long term use of an injection system from a source inside of containment. However, a total loss of containment heat removal scenario coupled with a containment isolation failure, including a pre-existing leak, is modeled to fail long term use of injection systems from a source inside of containment.

The information below will clarify how increasing the containment isolation failure probability in the model produces a change in CDF consistent with the ILRT extension assumptions. That is, the current pre-existing containment leakage probability discussed below is already set to the base case accident Class 3b value (2.3E-3) from the ILRT assessment. That value is increased by a factor of 5 to 0.0115 consistent with the assumed increase in Class 3b for the revised LERF determination. The impact of this change leads to the calculated increase in CDF as described.

Each event tree and associated success criteria and event tree documentation has been reviewed to assure that the ΔCDF result is appropriate.

<u>Explanation of Original QC PRA Modeling</u>

The QC FPIE Level 1 PRA model used to support the LAR includes loss of NPSH logic under gate NPSH-F. A pre-existing leak is one of many events that are modeled to impact NPSH with a loss of SPC and SDC. Gate NPSH-F is under gate ECCS-NPSH and may lead to loss of low pressure injection from Core Spray (CS) and RHR systems, if coupled with failure to provide containment heat removal early in the event, as shown in Figure 5-1.

Gate ECCS-NPSH, shown in Figure 5-2, fails Core Spray, and RHR pumps (LPCI Injection) unless Suppression Pool Cooling (SPC) or Shutdown Cooling (SDC) provides adequate heat removal early in the event. SDC is not credited where there is a high drywell pressure signal. A high drywell pressure signal is modeled to defeat SDC for the following events:

- Small, Medium and Large LOCA
- Single and Dual Unit LOOP
- Loss of Service Water

Failure of ECCS-NPSH fails gate CS-LPI which represents low pressure injection from the suppression pool. This logic and the impact of the ECCS-NPSH gate failing the CS-LPI gate, shown in Figure 5-3, is applied to almost all scenarios. An event tree mapping table found later in this response identifies any exceptions. Therefore, following a loss of SPC and SDC, a pre-existing containment leak fails CS and LPI from the suppression pool. Figure 5-3 also shows potential credit for FLEX/HCVS equipment, but this is currently not credited in the model for the ILRT application.

This failure logic is replicated for CS and LPI in Event Tree node QUV for post-containment failure via gate ZZ-FAIL (shown in Figure 5-4) and ZZ-VENT (shown in Figure 5-5). ZZ-VENT models successful suppression pool venting using the hardened vent. For successful venting, only sources outside the containment are credited.

Based on the modeling logic described above, the pre-existing leak fails CS and LPI injection pumps in all event trees. This was verified as resulting in core damage by quantifying the model with a pre-existing leak, no injection from outside the containment and no early SPC operation. Sequences from all initiators ended in core damage. As such, a total loss of containment heat removal scenario coupled with a containment isolation failure, including a pre-existing leak, fails long term use of injection systems from a source inside of containment.

A summary of initiating events and associated event trees is provided in Table 5-1.

Revised QC PRA Modeling

The QC FPIE PRA model used to support the LAR did not credit the potential for operators to align the RHR LPCI pumps and CS pumps to the CCSTs given the potential for loss of NPSH associated with the suppression pool. Additionally, the PRA model did not credit the potential for operators to refill the CCSTs to support long term injection. Procedures exist and operators have received training to support both actions.

To credit these operator actions two new basic events, each with a screening HEP value of 0.1 were added to the model.

- CCSTs Alignment - The basic event to credit the operator aligning the RHR Pumps or CS pumps to the CCSTs is added as human failure event HFE 1ECOP-CST-NP-H-- in the logic under gate ECCS-NPSH (presented in Figure 5-6). The action requires manual valve manipulations in the ECCS corner rooms of the respective pump. The new operator action is applicable prior to containment failure and is based on an existing HFE developed in view of aligning to the CCSTs post-containment failure. The HFE for post-containment failure is a developed as a detailed HEP, but is conservatively set to 1.0 in the model due to the potential for an adverse atmosphere in the Reactor Building following containment failure. The new detailed HEP for alignment prior to containment failure has a value less than 1E-2. For this risk assessment, the HEP is escalated to a screening value of 0.1 to address uncertainties such as the location of containment leakage under consideration. For sequences where this operator action is credited (i.e., loss of containment heat removal), core damage has not yet occurred. Leakage at 100La would equate to a hole in containment less of 3 in^2. Steam leakage through such a hole would be expected to have limited environmental impacts in the Reactor Building and should allow adequate access to the ECCS corner rooms.
- CCSTs Refill – The basic event to credit the operators aligning a refill source is added as 1ECOP-CSTFIL-H-- under the CCST-SUPPLY gate, and is applied only for DLOOP scenarios. The initial CCSTs inventory is not generally sufficient to support long term injection needs from both units simultaneously. Multiple means are available to provide

refill of the CCSTs, some of which require no offsite power (e.g., via diesel fire pumps, or via a portable pump). A significant amount of time is available to provide refill of the CCSTs for DLOOP scenarios. The HFE for refill was developed as a detailed HEP with a value of <1E-2. For this risk assessment, the HEP is escalated to a screening value of 0.1 to address uncertainties.

With the revised PRA modeling, the sensitivity to estimate the ΔCDF associated with increasing the containment isolation failure probability from 2.3E-3 to 1.15E-2 was re-performed. The CDF increase was 2.4E-8/yr (i.e., increased from a new base value of 2.39E-6/yr to 2.42E-6/yr). Thus with refined modeling, the ΔCDF estimate associated with potential containment overpressure impacts is significantly less than the 7.2E-8/yr value presented in the LAR.

It is additionally noted that the QC PRA model does not credit long term use of RCIC with loss of suppression pool cooling (such as would occur in an extended loss of AC power (ELAP). Such credit would further reduce overall CDF and reduce the impact of loss of NPSH to CS and LPCI. RCIC is capable of operating for greater than 24 hours without room cooling per room cooling calculation BSA-Q-97-04 (Rev 4). Industry studies such as the BWROG Fukushima Response Committee, BWROG-TP-14-018, Beyond Design Basis RCIC Elevated Temperature Functionality Assessment, December 2014, Revision 0 support long term use of RCIC. Incorporation of RCIC long term usage into an application specific PRA model has not been performed because of time constraints. A review of the top 10 NPSH sequence contributors associated with the model used for the LAR estimate of NPSH impacts found that approximately ~30% of the sequences involved credit for RCIC or HPCI for short term success (up to 8 hours), followed by depressurization. For such sequences, long term operation of RCIC would negate reliance on LPI or CS and would therefore reduce NPSH impacts even further.

Furthermore, it is noted that industry studies, such as plant tests at TVA Browns Ferry (e.g., NUREG/CR-2973) have shown that substantial NPSH margin exists for ECCS pumps of the vertical pump design used for BWR/3 and BWR/4 plants. Such tests have shown that the ECCS pumps can operate down to 60% of design NPSH. Such operational margin is not credited in this risk assessment.

A bounding approach to ΔLERF from the revised ΔCDF due to NPSH is included part of response to RAI 6.

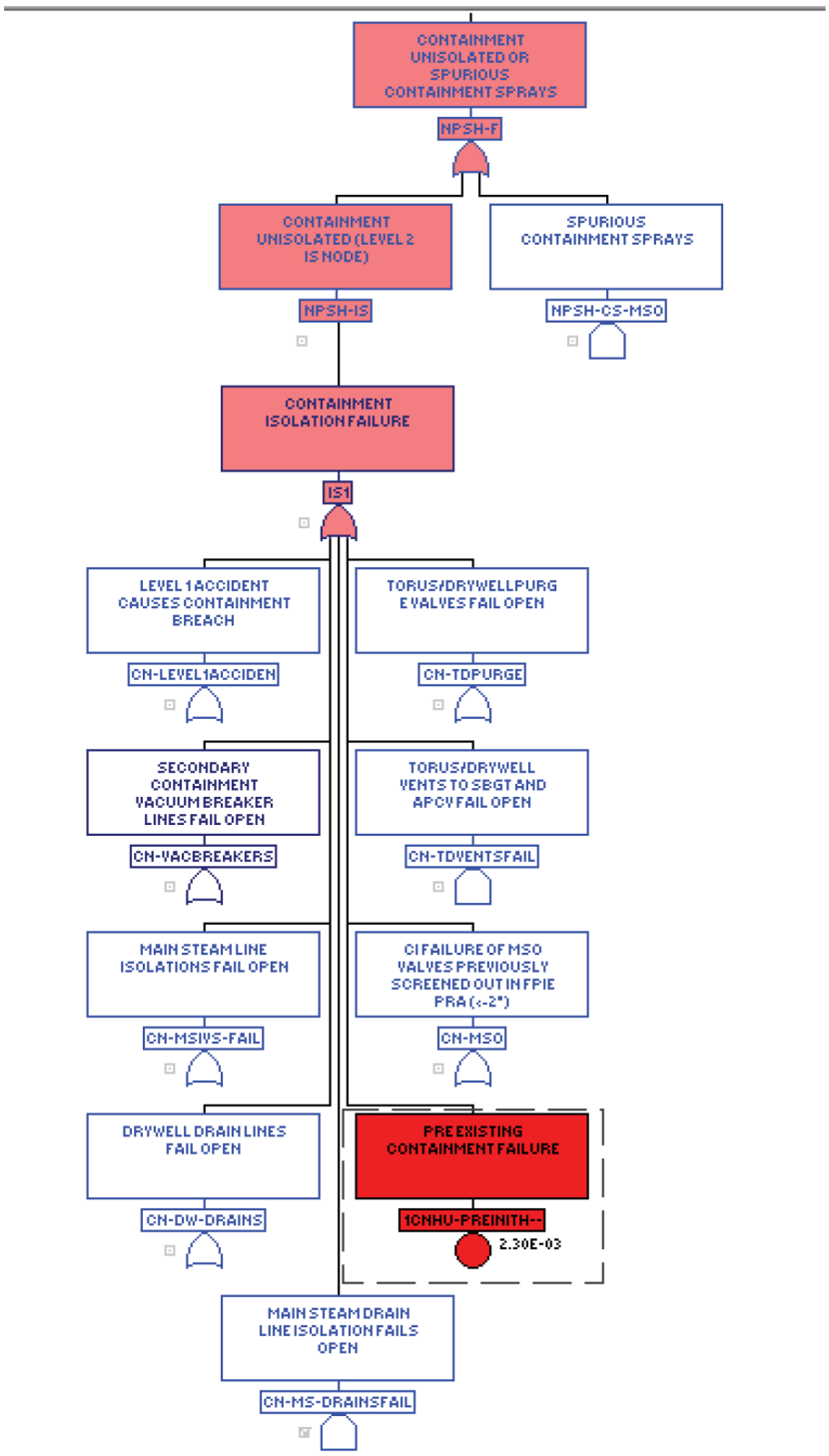

Figure 5-1
Pre-Existing Containment Failure Model Logic

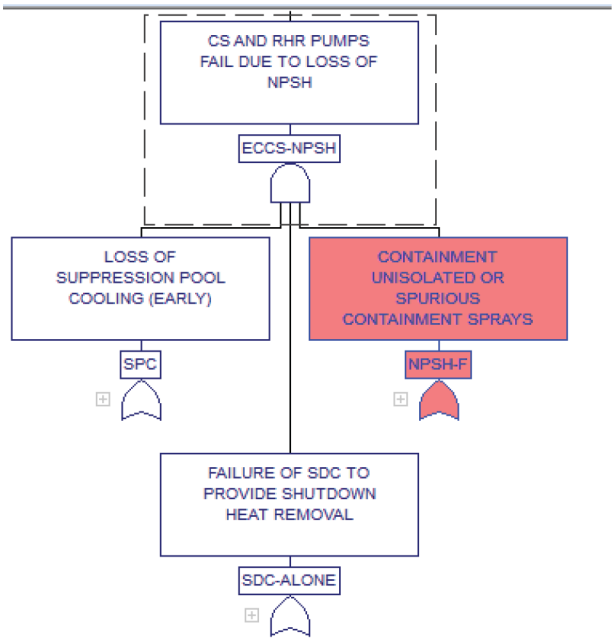

Figure 5-2
ECCS-NPSH Failure Logic

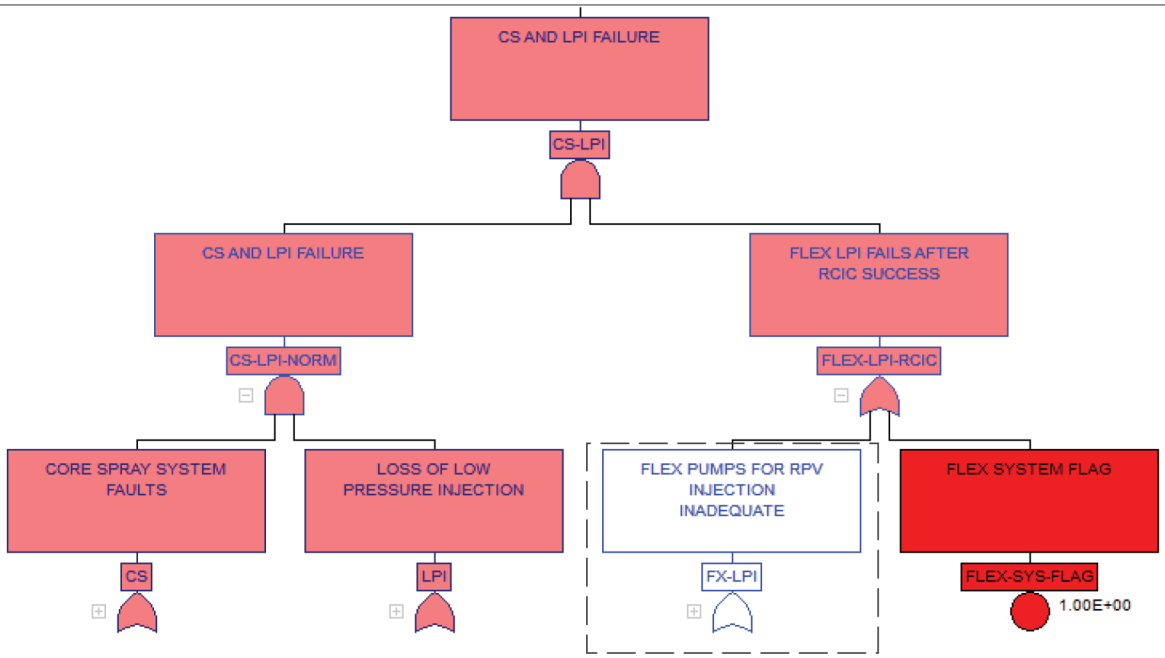

Figure 5-3
CS-LPI Model Logic

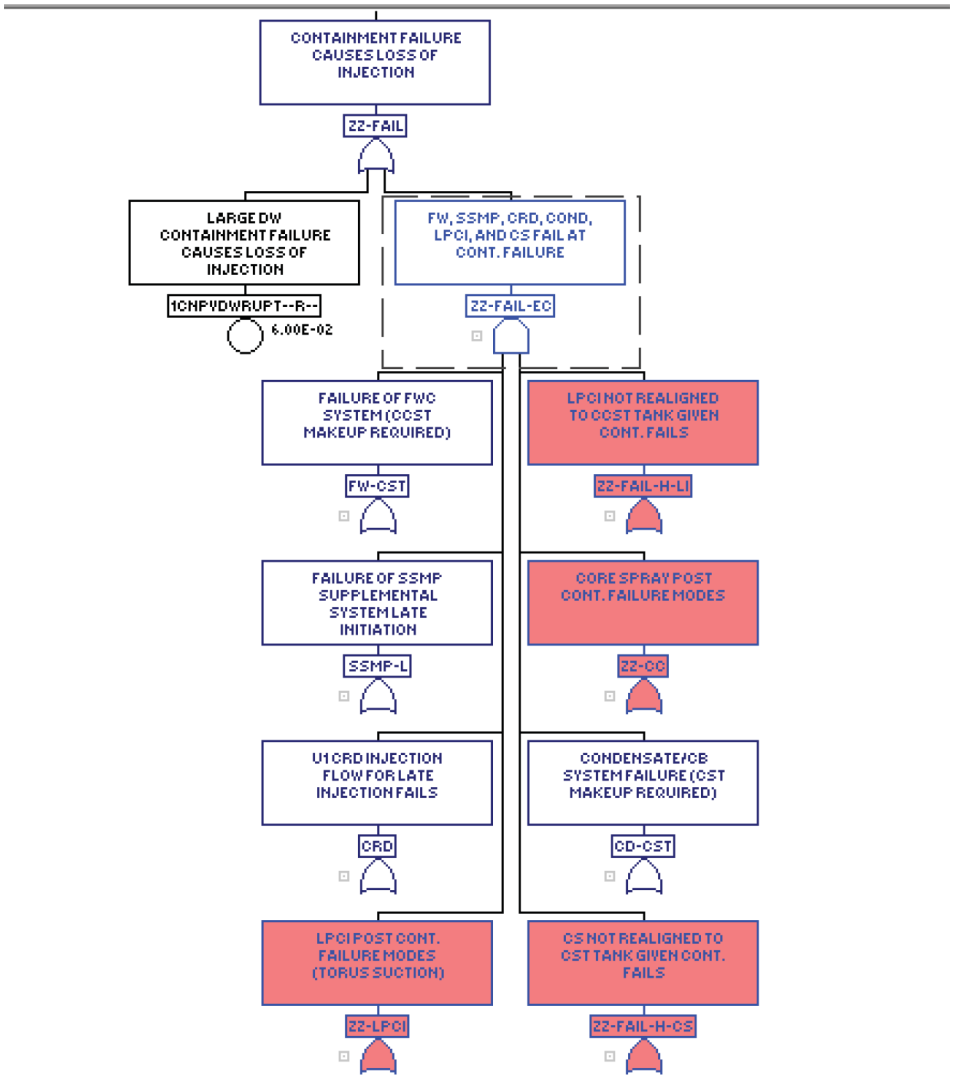

Figure 5-4
ECCS-NPSH Failure Impact to ZZ-FAIL

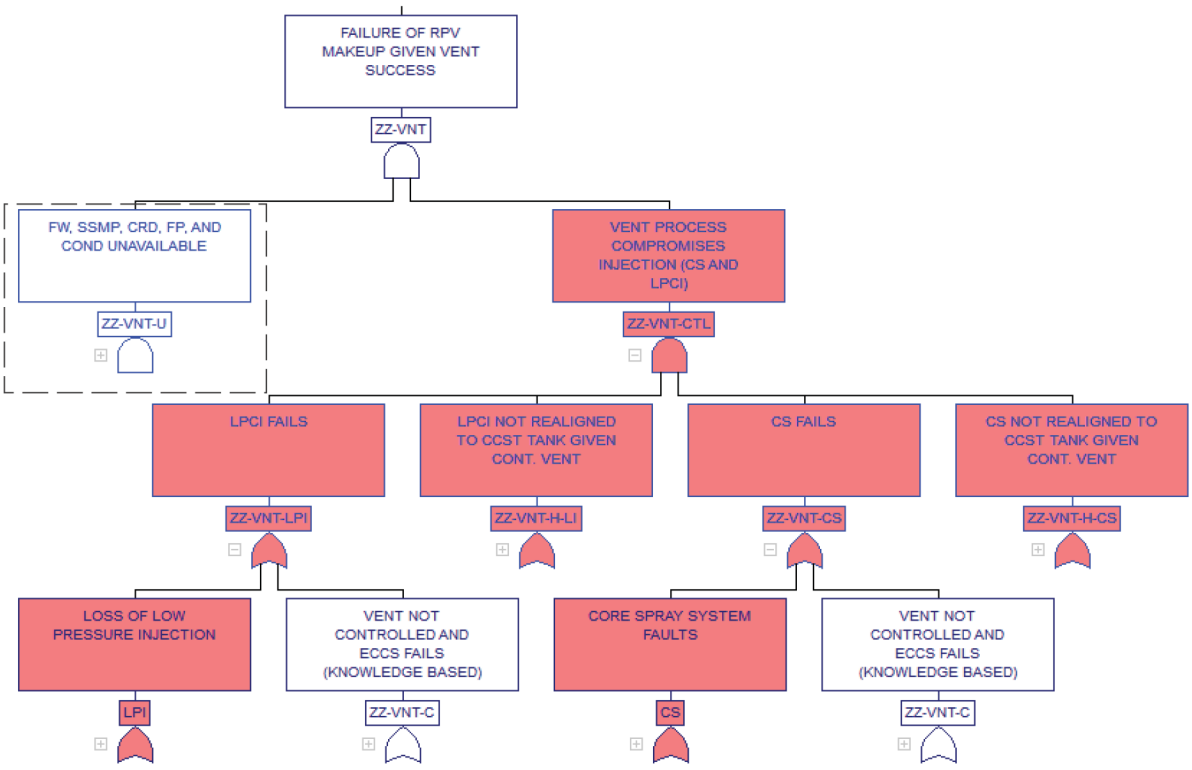

Figure 5-5
ECCS-NPSH Failure Impact to ZZ-VNT

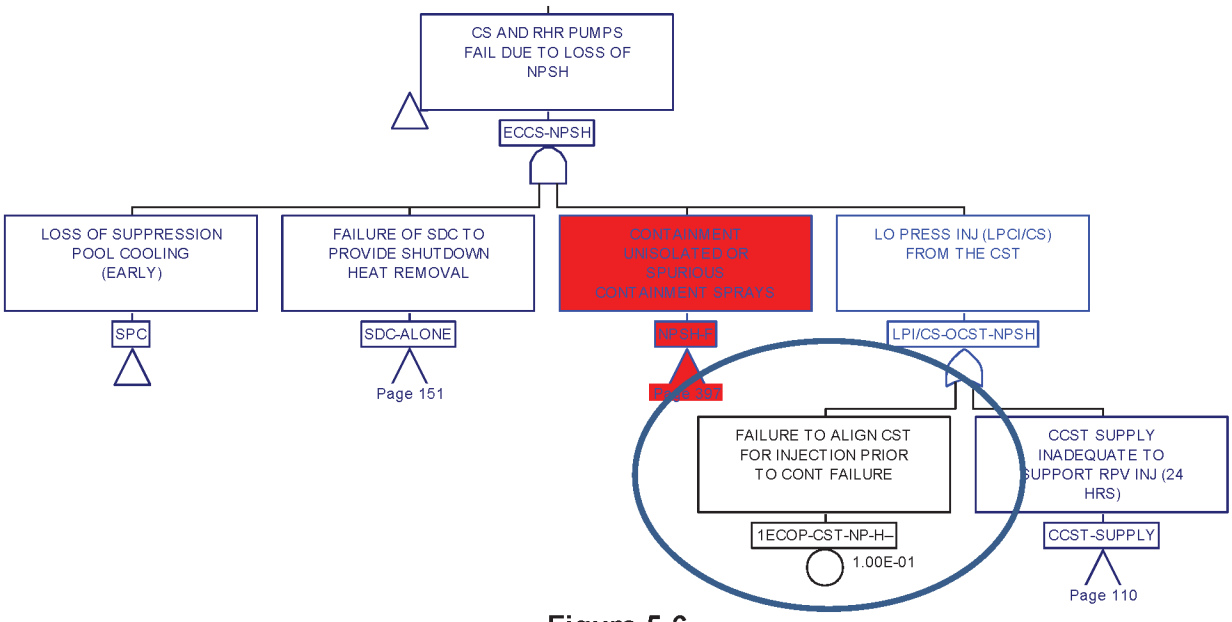

Figure 5-6
NEW OPERATOR ACTION LOGIC

TABLE 5-1

SUMMARY OF THE EVENT TREE MODELS

INITIATOR (EVENT TREE) [3]	CS-LPI FAILED WITH NPSH-ECCS TRUE	INJECTION SYSTEMS CREDITED WITH CONTAINMENT HEAT REMOVAL FAILURE[1][2]
TRANSIENT EVENT TREES		
Turbine Trip With Bypass (GTR)	Yes	CRD, CD/FW, SSMP
Loss of Feedwater (GTR)	Yes	CRD, CD, SSMP
MSIV Closure (GTR)	Yes	CRD, CD/FW, SSMP
Loss of Condenser Vacuum (GTR)	Yes	CRD, CD/FW, SSMP
Manual Shutdown (GTR)	Yes	CRD, CD/FW, SSMP
Internal Flood Initiators (GTR)	Yes	CRD, CD/FW, SSMP
Inadvertent Open Relief Valve (IORV)	Yes	CRD, CD/FW, SSMP
SORV (IORV)	Yes	CRD, CD/FW, SSMP
Single Unit Loss of Offsite Power (LOOP)	Yes	CRD (from opposite unit), SSMP
Dual Unit Loss of Offsite Power (DLOOP)	Yes	SSMP
LOOP/DLOOP and SORV (SORV-LOOP and SORV-DLOOP)	Yes	CRD (LOOP only), SSMP
LOCA EVENT TREES		
Small Break Loss of Coolant Accident (LOCA) (SLOC-S, Steam and SLOC-W, Water)	Yes	CD, SSMP
Medium Break Loss of Coolant Accident (MLOCA-S, Steam and MLOCA-W Water)	Yes	CD, CRD, SSMP CRD and SSMP credited if HPCI is initially available. CRD and SSMP not credited for Water LOCA. CD must have SBCS(4) make-up for Water LOCA.
Large Break Loss of Coolant Accident (LOCA) (Includes Steam and Liquid breaks > 0.2 ft^2. Inputs to Steam LOCA event tree include multiple SORVs and Inadvertent ADS) (LLOC-S, Steam and LLOC-W, Water)	Yes	SSMP, CD SSMP not credited for Water LOCA. CD must have SBCS makeup for Water LOCA.
LOCAs Outside Containment (BOC and BOC-TB Unisolated Break in Turbine Building)	Yes	CD CD not credited for BOC-TB
Interfacing Systems LOCA (ILOCA)	Yes	CD CD must be supported by SBCS
Excessive LOCA (ELOCA)	Yes	SSMP SSMP is only credited if SPC is successful late
ATWS EVENT TREES		
Failure to Scram Transient Events With Main Condenser (MC) Available (AT-MC-AF)	N/A (CS and LPI not credited in this event tree).	Feedwater injection.
Failure to Scram Transient Events With Main Condenser Not Available (AT-MC-FL)	Yes	SSMP with Make-up from CCST and TDV for heat removal (Footnotes

TABLE 5-1

SUMMARY OF THE EVENT TREE MODELS

INITIATOR (EVENT TREE) [3]	CS-LPI FAILED WITH NPSH-ECCS TRUE	INJECTION SYSTEMS CREDITED WITH CONTAINMENT HEAT REMOVAL FAILURE[1][2]
		5 and 6 apply)
Transient, LOOP, DLOOP with an SORV Plus a Failure to Scram (Also includes IORV initiating event scenario) (AT-SORV, LOOP and AT-SORVD)	Yes	SSMP with Make-up from CCST and TDV for heat removal (Footnotes 5 and 6 apply)
LOOP/DLOOP ATWS (ATL-MCFL, LOOP and ATD-MCFL, DLOOP)	Yes	SSMP with Make-up from CCST and TDV for heat removal (Footnotes 5 and 6 apply)
SPECIAL INITIATOR EVENT TREES[7]		
Loss of Two 125V DC Buses Loss of SW Loss of Instrument Air Loss of Nitrogen Reactor Water Level Reference Leg Failure Loss of Drywell cooling Loss of TBCCW Loss of Bus 11 Loss of Bus 12 Loss of Bus 13 Loss of Bus 14 Loss of Bus 18 Loss of MCC 18-2 Torus or Connected Pipe Rupture Non-SW, Non-Torus Pipe Rupture in Torus Area Service Water Rupture in Reactor Building on Elevation 595' or above DGCW or Clean Demin Rupture in Reactor Building on Elevation 595' or above Service Water Rupture in Turbine Building on Elevation 595' or above Pipe Spray of Buses 13 and 14 Service Water Rupture in a Corner Room (Not applicable) FPS, Fuel Pool, or CCST Pipe Rupture in Reactor Building on Elevation 595' or above Cable Spreading Room MCCs Flooded Service Water Pipe Rupture in HPCI Compartment Pipe Spray of MCCs 18-1 and 19-4 Miscellaneous Spray Effects on Electrical Buses FPS, DGCW, or RHRSW Ruptures in the Turbine Building at Elevation 595' or above	See General Transient Event Tree	See General Transient Event Tree

Notes to Table 5-1:

(1) CRD and SSMP Injection sources are the Clean Condensate Storage Tanks (CCST). The CD and FW injection sources are the Hotwell with make-up from the CCSTs (A&B).

(2) CRD requires early HPCI or RCIC success.

(3) An initiator will transfer to another event tree if a consequential event, such as SORV, DLOOP or ATWS occurs.

(4) SBCS is an acronym for Standby Coolant Supply. SBCS takes suction from the ultimate heat sink using Service Water Pumps and provides make-up to the hotwell.

(5) Containment venting may be initiated after other means of containment heat removal (e.g., suppression pool cooling, PCS) are unavailable. Containment venting directs the release of steam to the containment venting pathways in a controlled manner and allows the steam to be scrubbed by the suppression pool water if the wetwell pathway is used. Venting is judged a feasible heat removal method if the operator has been successful in controlling the ATWS scenario except for establishing containment heat removal. The containment venting system will be necessary to control containment pressure earlier in an ATWS sequence than for a general transient.

(6) If the ATWS has been successfully mitigated using SBLC and adequate RPV injection is available, then venting the containment can provide adequate containment control. However, the available systems for RPV injection are reduced to those that take suction from outside containment.

(7) Special Initiators are treated in the General Transient Event Tree unless consequential events occur resulting in transfer one of the event trees listed below. Exception is the Total Loss of 125 VDC due to low initiating event frequency. Consequential events transfer to one of the following trees:

- SORV Event Tree
- Large LOCA Event Tree
- Failure to Scram Event Tree
- Failure to Scram Event Tree with an SORV

NRC RAI 6

6. Subsequent to resolving the requests above, confirm that the total change in LERF and total LERF are still within the "small" risk increase regions of Regulatory Guide 1.174. If necessary, re-perform the calculations in the LAR using any updated CDF or LERF values provided in response to the requests above.

Response to RAI 6:

Based on responses to RAIs 1c, 3b, and 5c, the total change in LERF and total LERF are recalculated and shown to be within the "small" risk increase regions of Regulatory Guide 1.174. Sections of 5.7 from Attachment 3 of the LAR are reproduced and updated as follows:

Revised (Updated) Portions of Section 5.7

QCNPS Fire Risk Discussion

LERF was not quantified in the IPEEE; therefore a LERF estimate must also be developed. The internal events LERF value for the QC FPIE model is **2.10E-07/yr. (per RAI Response 1c)**. Consistent with the approach for CDF, the QC Fire LERF is assumed to be a factor of **8.7 (per RAI Response 3b)** higher than the FPIE model LERF.

Fire LERF = FPIE LERF (**2.10E-07 /yr.**) x **8.7** (Fire CDF w/o Class II non-LERF contributions) ÷ FPIE CDF (w/o Class II non-LERF contributions))
Fire LERF = **1.83E-06/yr**.

Note that the FPIE LERF value of **2.10E-07/yr.** includes the Class II LERF contribution (i.e., 4.17%) and therefore, the Fire LERF estimate developed by this ratio approach also includes the Class II contribution from a release with late declaration of a GE. The FPIE LERF value does not include Class II CDF with early declaration of a GE (i.e., 95.8%). Therefore, the approach used above is appropriate for estimating a Fire LERF value (i.e., Class II contributions do not need to further be subtracted).

QCNPS Seismic Risk Discussion

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

For CDF, the seismic CDF is a factor of 9.25 greater than FPIE CDF (i.e., 2.7E-05 / 2.92E-06). Given this, it is reasonable to assume that the total CDF impact from seismic risk can be approximated by assuming a factor of 9.25 additional contribution to CDF compared to the internal events evaluation alone. Using a base FPIE LERF value of **2.10E-07**/yr and multiplying by 9.25 for seismic, gives a total LERF estimate for the seismic PRA model of **1.94E-06/yr**.

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

External Events Impact Summary

In summary, the seismic and fire CDF values described above result in an external events bounding risk estimate of **9.83E-05/yr**. Seismic and Fire LERF values derived from CDF values in sections 5.7.2 and 5.7.3 and as shown in Table 5.7-3 below sum to LERF value of **3.77E-06/y**r., which is **18.0** times higher than the internal events LERF.

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

TABLE 5.7-3 (UPDATED)

QCNPS EXTERNAL EVENTS CONTRIBUTOR SUMMARY

EXTERNAL EVENT INITIATOR GROUP	CDF (1/YR)	LERF (1/YR)
Fire	**7.13E-05**	**1.83E-06**
Seismic	**2.7E-05**	**1.94E-06**
High Winds	Screened	Screened
Other Hazards	Screened	Screened
Total For External Events (for initiators with CDF/LERF available)	**9.83E-05**	**3.77E-06**
Internal Events	**2.92E-06**	**2.10E-07**

It is noted that the total CDF in Table 5.7-3 for internal and external events when combined totals 1.01E-04/yr, slightly exceeding the 1.0E-04/yr threshold. RG 1.174 notes the following:

> When the calculated increase in CDF if very small, which is taken as being less than 10^{-6} per reactor year, the change will be considered regardless of whether there is a calculation of the total CDF (Region III). While there is no requirement to calculate the total CDF, if there is an indication that the CDF may be considerably higher than 10^{-4} per reactor year, the focus should be on finding ways to decrease rather than increase it.

Extending the ILRT interval primarily results in a potential impact to LERF, however plants which credit NPSH for ECCS success, such as Quad Cities, may have a calculated slight increase in CDF. This potential increase was conservatively estimated as 2.4E-08/yr, which is well below the 10^{-6} CDF threshold identified in RG 1.174. With respect to reducing risk, Quad Cities continues to implement plant changes that reduce CDF and LERF. Next year the plant is scheduled to complete additional modifications for the Hardened Containment Vent System (HCVS) that will meaningfully reduce risk, as discussed later in this response. It is also noted that the Fire and Seismic CDF results used in this ILRT assessment are considered conservative representations of risks for these hazards.

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

TABLE 5.7-4 (UPDATED)

QC EXTERNAL EVENTS TO INTERNAL EVENTS CDF COMPARISON

EVENT INITIATOR GROUP	CDF (1/YR)	ADJUSTED CDF (1/YR)	EVENT INITIATOR GROUP	ADJUSTED CDF (1/YR)	INITIAL MULTIPLIER	APPLICABLE MULTIPLIER PORTION(2)
Fire	**7.13E-5**	**1.69E-5(1)**	FPIE (reduced Class II frequency)	1.94E-6(1)	**8.7**	**5.8**
Seismic	2.7E-5	N/A	FPIE (full Class II frequency)	2.92E-6	9.3	9.3
External Event CDF to FPIE CDF Multipler					**15.1**	

Note to Table 5.7-4:

(1) 95% of Class II CDF contribution removed from Fire CDF as discussed previously, and also the Full Power Internal Events CDF to develop a multiplier for Fire CDF/FPIE CDF. **(See also response to RAI 3c for fire risk)**

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

External Events Impact on ILRT Extension Assessment

The EPRI Category 3b frequency for the 3-per-10 year, 1-per-10 year, and 1-per-15 year ILRT intervals are shown in Table 5.6-1 as 6.89E-09/yr, 2.37E-08/yr, and 3.70E-08/yr, respectively. Using an external events LERF multiplier of **15.1** (multiplier from Table 5.7-4) for QCNPS, the change in the LERF risk measure due to extending the ILRT from 3-per-10 years to 1-per-15 years, including both internal and external hazards risk, is estimated as shown in Table 5.7-5.

TABLE 5.7-5 (UPDATED)

QCNPS 3B (LERF/YR) AS A FUNCTION OF ILRT FREQUENCY FOR INTERNAL AND EXTERNAL EVENTS (INCLUDING AGE ADJUSTED STEEL LINER CORROSION LIKELIHOOD)

	3B FREQUENCY (3-PER-10 YR ILRT)	3B FREQUENCY (1-PER-10 YEAR ILRT)	3B FREQUENCY (1-PER-15 YEAR ILRT)	LERF INCREASE[1]
Internal Events Contribution	6.9E-09	2.4E-08	3.7E-08	3.0E-08
External Events Contribution (Internal Events x **15.1**)	**1.0E-07**	**3.6E-07**	**5.6E-07**	**4.6E-07**
Combined (Internal + External) (RAIs 1c and 3b)	**1.1E-07**	**3.8E-07**	**6.0E-07**	**4.9E-07**
NPSH Internal Events Contribution (RAI 5)	--	--	--	**2.4E-08**
NPSH External Events Contribution (Internal Events x 15.1) (RAI 5)	--	--	--	**3.6E-07**
NPSH Combined (Internal + External) (RAI 5)	--	--	--	**3.8E-07**
Total Combined (Internal + External) (RAIs 1c, 3b, 5)	--	--	--	**8.7E-07**

Note to Table 5.7-5:

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

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

TABLE 5.7-6

COMPARISON TO ACCEPTANCE CRITERIA INCLUDING EXTERNAL EVENTS CONTRIBUTION FOR QCNPS

CONTRIBUTOR	ΔLERF	ΔPERSON-REM/YR	ΔCCFP[1]
Internal Events	3.0E-08	1.0E-02 (0.31%)	1.03%
External Events	**4.6E-07**	1.5E-01[2] (0.31%)	1.03%
NPSH (Int + Ext)	**3.8E-07**	**1.3E-01[3] (0.31%)**	**1.03%**
Total	**8.7E-07**	**2.9E-01 (0.31%)**	**1.03%**
Acceptance Criteria	<1.0E-6/yr (“small”)	<1.0 person-rem/yr or <1.0%	<1.5%

Notes to Table 5.7-6:

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

(2) Calculated as the FPIE value times the external events multiplier of **15.1** developed in Table 5.7-4.

(3) Calculated as the FPIE value times a multiplier of 12.7 (i.e., 3.8E-07/yr / 3.0E-08/yr)

The **8.7E-07/yr** increase in LERF due to the combined internal and external events from extending the ILRT frequency from 3-per-10 years to 1-per-15 years falls within Region II between 1.0E-7 to 1.0E-6 per reactor year (“small” change in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the “small” change range, the risk assessment must also reasonably show that the total LERF is less than 1.0E-5/yr. Similar bounding assumptions regarding the external event contributions that were made above are used for the total LERF estimate.

From Table 4.2-2, the LERF (High Early) due to postulated internal event accidents is **2.1E-07/yr** for QCNPS. As discussed in Sections 5.7.1 and shown in Table 5.7-3, the LERF estimate for the Fire PRA model is **1.8E-06 /yr.** As discussed in Sections 5.7.2 and shown in Table 5.7-3, the total LERF estimate for the Seismic PRA model is **1.9E-06 /yr**. The total LERF values for QCNPS are shown in Table 5.7-7.

TABLE 5.7-7 (UPDATED)

IMPACT OF 15-YR ILRT EXTENSION ON LERF FOR QCNPS

LERF CONTRIBUTOR	(1/YR)
Internal Events LERF	**2.1E-07**
Fire LERF	**1.8E-06**
Seismic LERF	**1.9E-06**
Internal Events LERF due to ILRT (at 15 years) (1)	3.7E-08
External Events LERF due to ILRT (at 15 years) (1)	**5.6E-07** [Internal Events LERF due to ILRT * 15.1]
NPSH LERF (2)	**3.8E-07**
Total	**4.9E-06/yr**

Note to Table 5.7-7:

(1) Including age adjusted steel liner corrosion likelihood as reported in Table 5.7-5.
(2) NPSH LERF is assumed to be equal to NPSH delta CDF for internal and external events as reported in Table 5.7-6.

As can be seen, the estimated upper bound LERF for QCNPS is estimated as **4.9E-06/yr.** This value is less than the RG 1.174 requirement to demonstrate that the total LERF due to internal and external events is less than 1.0E-05/yr.

Conclusion

The above calculations demonstrate that the quantitative evaluations associated with responses to RAIs 1c, 3b, and 5c do not change the conclusions of the ILRT risk assessment.

It is additionally noted that the QC ILRT Risk Assessment has a number of conservatisms that are not reflected in the quantified risk metrics. These include the following:

- FLEX/HCVS – Quad Cities will complete the implementation of FLEX/HCVS equipment and training in 2018 to respond to beyond design basis events. As shown in Figure 5-3, modeling for FLEX/HCVS components exist in the PRA model used to evaluate the ILRT interval extension, however, FLEX/HCVS was not credited in the ILRT evaluation. The HEPs included in the FLEX/HCVS modeling are conservative screening values. Therefore, the model is judged to present a conservative (i.e., less than full) credit for FLEX/HCVS for CDF and LERF reduction. For sensitivity purposes, credit for FLEX/HCVS reduces the FPIE CDF and LERF as follows:

Metric	Base Case Result (/yr)	FLEX/HCVS Credit (/yr)	Difference
CDF	2.92E-06	1.59E-06	-45.5%
LERF	1.97E-07	1.53E-07	-22.3%

- Inerted Containment – Quad Cities utilizes an inerted containment design. The pre-existing failure utilized in the EPRI ILRT methodology models a pre-existing leak that is 100 times the plant technical specification leakage. A containment leak of this magnitude would require a flow area of approximately 2.8 in^2. A pre-existing leak of this size would be readily detectable based on makeup required to maintain the inerted atmosphere. Therefore the probability of a pre-existing containment failure existing

without detection, as postulated using the EPRI methodology, is considered to be very low.

RAI 7

Background

Title 10 of the *Code of Federal Regulations* (10 CFR) 50, Appendix J, Option B, requires a general visual inspection of the containment prior to each Type A test and at a periodic interval between tests. Nuclear Energy Institute 94-01, Revision 3-A, recommends these inspections be performed in conjunction with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWE, required examinations. Section 3.5.2 of the LAR indicates that Quad Cities Nuclear Power Station (QCNPS) will use the required ASME Code examinations to meet the Appendix J requirement regarding visual inspections.

The regulation at 10 CFR 50.55a(b)(2)(ix)(A) imposes a condition on the use of ASME Code, Section XI, Subsection IWE, which requires licensees to evaluate the acceptability of inaccessible metal containment areas when conditions exist in accessible areas that could indicate the presence of, or result in degradation to, such inaccessible areas.

Issue

Table 3.5-3 of the LAR summarizes the results of the ASME Code, Section XI, Subsection IWE, metal containment inspections performed over the last three outages. The table notes that both units have identified recordable indications on the moisture barrier; however, no discussion is provided regarding the containment behind the moisture barrier. A degraded moisture barrier could indicate the presence of degradation in the inaccessible areas behind the moisture barrier.

It is not clear to the NRC staff how EGC determined that the degraded moisture barrier did not allow moisture to contact the containment and cause degradation in an inaccessible area.

Request

Summarize the results of any 10 CFR 50.55a(b)(2)(ix)(A) evaluations associated with noted moisture barrier indications, or explain how EGC determined that the inaccessible portions of the containment were not impacted and that an evaluation was not necessary.

Response

Based on the review of work orders that were performed during Refueling Outages Q1R23 and Q2R22, the containment liner at the degraded airlock moisture barrier was inspected after the small degraded sections of the seals were removed. No degradation of the containment liners behind the degraded moisture barriers was identified, and the integrity of the moisture barriers was restored to an acceptable condition. Additional information regarding the program to monitor potential drywell degradation is provided in Attachment 1, Section 3.6.5, of the LAR.

RAI 8

Background

Appendix J to 10 CFR 50, Option B, requires a general visual inspection of the containment prior to each Type A test and at a periodic interval between tests for indications of structural deterioration that may impact leak-tightness. Section 3.6.5 of the LAR notes that inspections of the sand pocket region and drywell liner area are conducted at QCNPS during each refueling outage. Table 3.6.5-1 of the LAR summarizes the results of the last two inspections.

Issue

Table 3.6.5-1 Note (2) of the LAR, related to Unit 1, states that there is leakage but that the leakage is not from the drains and appears to be groundwater leakage with no structural impact. Note (4), related to Unit 2, indicate issues with groundwater leakage. From the notes, it appears the leakage is an ongoing issue.

It is not clear to the staff how EGC determined that the leakage is groundwater and that the leakage is not impacting the structural integrity or leak-tightness of the containment.

Request

For both units, explain how EGC determined that the identified leakage is groundwater and that the leakage is not impacting the structural integrity or leak-tightness of the containment.

Response

The term "ground water" was unfortunately used in the initial corrective action documents. Results of the analysis of the water was generally inconclusive; however, the likely source of the water is refueling outage fuel pool water leakage past various seals. This conclusion is based on the results of one of three chemistry samples that were conclusive. The other samples were not of sufficient volume and/or the chemical constituents of the sample did not point to a clear source for the water.

Procedure QCTS 0820-11, "Surveillance of Dryer-Separator Pool, Spend Fuel Pool, and Drywell Liner Drains," provides instruction for inspection of Drywell Liner Sand Pocket Drain Lines for water leakage. The procedure also requires inspection of the water leakage through the Dryer-Separator Pit and Spent Fuel Pool liners. The procedure provides for initiation of a Corrective Action Program issue report (IR) if leakage is detected during the inspection. The inspection is performed at least once per operating cycle with the dryer-separator pool flooded. Preventive maintenance tasks have been established to track completion of both the validation that the drains are open, and to check for leakage after cavity flood up during refueling outages. Additional information regarding the program to monitor potential drywell degradation is provided in Attachment 1, Section 3.6.5, of the LAR.