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RS-17-130

September 28, 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 (RS-17-051) (CAC. Nos. MF9675 and MF9676)," dated September 19, 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 Reference 2. In response to this request, EGC is providing the attached information.

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 28th day of September 2017.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Response to Request for Additional Information
2. QC-LAR-05, "Quad Cities ILRT RAI Round 2 Response"

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

RAI 5-A

Section 4.2.6 of EPRI [Electric Power Research Institute] TR [Topical Report)-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 [core damage frequency]," therefore requiring a risk assessment. In response to request for additional information (RAI) 5, EGC described the probabilistic risk assessment (PRA) modeling to estimate the change in risk from loss of NPSH to the ECCS pumps and provided a ΔCDF estimate of 2.4E-8/year.

a. As described in the RAI response, the risk analysis appears to assume a total loss of containment heat removal for all accident scenarios that were considered, such as transients, loss-of-coolant accident, anticipated transient without scram, special initiators, etc., which are listed in Table 5-1 of the RAI response. However no justification was provided for scenarios with containment decay heat removal available. Explain and justify why the loss of containment overpressure impacting NPSH for the ECCS injection is not a concern in any accident scenario with containment decay heat removal available. Alternatively, if any additional accident scenarios are identified to contribute to the risk increase, provide an updated estimate of ΔCDF.

b. The RAI response attempts to explain the PRA modeling for scenarios "post containment failure" and for scenarios with "successful pool venting" in Figures 5-4 and 5-5. Since the containment would already be failed due to the postulated pre-existing containment leak, further PRA modeling appears unnecessary and the application of the model described in Figure 5-1 to those post containment failure scenarios may result in a reduction in the risk estimate. The RAI response also states that for "successful pool venting," only sources outside the containment are credited. Clarify how the PRA model described in the response to RAI 5 correctly captures the risk impact from accident scenarios "post containment failure" and those with "successful pool venting."

c. In response to RAI 5.c, ΔLERF (large early release frequency) resulting from loss of NPSH was equated to ΔCDF. If a new method to estimate ΔLERF is deemed necessary in response to items a or b above, describe and justify any credit taken for reducing ΔLERF below the value for ΔCDF.

Response

Response is provided in Attachment 2.

RAI 8-A

Background

RAI 8 requested that EGC explain how it determined that the identified leakage near the containment structure was groundwater and is not impacting the structural integrity or leak-tightness of the containment.

In its response, EGC explained the leakage was not groundwater but did not address how it was determined the leakage was not impacting containment. The response contained no information regarding the leakage location relative to containment, and no discussion of whether or not the leakage is contained within the sand pocket drain lines.

Issue

If the leakage is contained within the sand pocket drain lines (i.e., no indications of leakage through concrete in the general vicinity of the drains), the NRC staff understands how it can be determined the leakage is not impacting the containment. However, if leakage is occurring through concrete in the general area of the drains, the leakage could be impacting inaccessible portions of containment. The response did not provide enough information for the NRC staff to determine that the leakage is contained within the drains, or if the leakage is outside the drain lines (as implied in Notes 2 and 4) that the leakage is not impacting containment.

Request

For both units, explain how EGC determined that the identified leakage is not impacting the structural integrity or leak-tightness of the containment. The explanation should include whether or not the leakage is contained such that it is not impacting containment and how it was determined that the leakage is not impacting inaccessible portions of the containment (e.g., location of leakage precludes it from contacting containment, minimal amount of leakage or sporadic nature of leakage limits possibility of corrosion).

Response

The historical leakage that Notes 2 and 4 refer to are attributed to containment liner sand pocket drain leakage and drywell pedestal wall leakage. Each of these locations is addressed separately below.

Containment Liner Sand Pocket Drain Leakage

Each Quad Cities Nuclear Power Station (QCNPS) unit is designed with a sand pocket drain system. The purpose of this system is to provide a flow path to allow drainage of moisture that may be introduced into the annulus (i.e., expansion gap) between the containment and the primary containment shield wall. Water can be introduced from leakage of the refuel cavity past the refueling bellows drain line expansion joints during refueling or because of the introduction of water at other drywell penetrations. This water migrates to the sand pocket and then passes through the sand pocket drain lines. The annulus and sand pocket encircle the entire containment. There are four sand pocket drain lines on each QCNPS unit that drain moisture from the sand pocket to the drywell pedestal wall, and preventive maintenance tasks are in place to periodically validate (i.e., every refueling outage) that the sand pocket drain lines are not clogged. A moisture seal at the bottom of the sand pocket prevents leakage from reaching the inaccessible area of the containment liner.

Reactor cavity flood up during refueling outages, or maintenance outages that involve movement of fuel between the reactor core and spent fuel pool, is the most likely source of leakage that could impact the containment liner. This water (i.e., fuel pool water) is very pure and noncorrosive to the containment liner. As described in Section 6.2.1.2.1.2, "Drywell Corrosion Potential," of the QCNPS Updated Final Safety Analysis Report (UFSAR), water present in the sand pocket is noncorrosive to the containment liner.

For Unit 1, no sand pocket drain leakage has been detected. One drain line is known to be plugged. However, as discussed above, the design of the sand pocket drain system includes multiple drain lines. Therefore, leakage that collects in the area of the sand pocket drain system near the plugged drain line would pass on to an adjacent drain.

For Unit 2, sand pocket drain leakage was previously identified in 2010 and 2016. Results of a chemical analysis determined that the likely source of the water was refueling outage fuel pool water leakage. This leakage is observed for a few hours during reactor cavity flood up and then stops, which further supports the likely source being fuel pool water. As described above, fuel pool water is very pure and noncorrosive to the containment liner. In addition, the design of the sand pocket drain system ensures that this water is contained and diverted away from the inaccessible containment liner area.

In summary, the design of the sand pocket drain system, including the moisture seal, precludes leakage from contacting the inaccessible containment liner. Therefore, there is no suspected leakage impact on the Unit 1 and Unit 2 containment liner structural integrity or on the leak-tightness of the containment liner from above the sand pocket drain system.

<u>Drywell Pedestal Wall Leakage</u>

Damp areas identified as leakage have been observed on the Unit 1 and Unit 2 drywell pedestal walls. Some of the leakage has historically been observed around the sand pocket drainage outlets and down the pedestal wall, and in bays other than where the sand pocket drains are located. The appearance of the leakage is sporadic and not active. Observers of the wall leakage have attributed the source to ground water; however, the observed wall leakage is of insufficient quantity to support chemical analysis. The source of the moisture has not been conclusively identified.

The wall leakage could potentially be ground water. However, as described in Section 6.2.1.2.1.2 of the UFSAR, the hydraulic pressure of the water table is not sufficient for ground water migration to reach the inaccessible area of the containment liner. Specifically, the elevation of the bottom of the drywell is 569 feet 10 inches. The normal ground water level is slightly higher than the pool stage of the Mississippi River (i.e., 572 feet 0 inches), resulting in a negligible driving head of approximately 4 feet.

<u>Conclusion</u>

The minimal amount of leakage observed and the sporadic nature of leakage limits the possibility of corrosion. The drywell steel is protected against corrosion by a 2-mil thick inorganic zinc-filled coating and is embedded in concrete 19 feet 10 inches above the rock surface. The concrete plug under the drywell is designed for a thermal gradient of 100°F, from an operating temperature in the drywell of 150°F to a temperature at the rock interface of 50°F. The thermal stress in the concrete of 572 psi is greater than the conservative value of 450 psi at which concrete would crack; therefore, cracking as normally expected with any concrete structure could occur. However, the heavily reinforced concrete plug would inhibit crack propagation and, in fact, would not permit a thermally-induced crack to open wide enough to act as a water passage. With all these positive factors (i.e., protective coating, negligible driving head for water intrusion, low thermal stress which will not develop a continuous crack in the concrete, and the heavily reinforced concrete plug), the potential for corrosion of the drywell is practically nonexistent. The expansion gap has provisions for drainage of moisture into the basement of the reactor building by means of a sand pocket and drain tube arrangement at the bottom of this space. In addition, an extensive review of the potential for drywell steel corrosion in the area of the sand pocket was performed, in response to NRC Information Notice 86-99 and Generic Letter 87-05, that concluded the water present in the sand pocket or inside the drywell was noncorrosive (i.e., based on testing), and there was no evidence of apparent corrosion. Therefore, the observed sand pocket drain leakage and drywell pedestal wall leakage does not impact the structural integrity of the containment liner or leak tightness of the containment liner.

ATTACHMENT 2

QC-LAR-05, "Quad Cities ILRT RAI Round 2 Response"

Exelon Generation®

RISK MANAGEMENT TEAM

RM DOCUMENTATION NO. QC-LAR-05 **REV:** 0 **PAGE NO.** 1

STATION: Quad Cities Nuclear Power Station (QCNPS)

UNIT(S) AFFECTED: 1 and 2

TITLE: Quad Cities ILRT RAI Round 2 Response

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

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: MAAP data files QC ILRT MAAP RUNS – 9-26.zip, 4.156 KB, dated 9/20/2017 2:01 PM, located on JH Campbell server: W:\Engineer\Exelon\Quad\Applications\2016_ILRT\2017_Rnd 2 RAI Files

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

This RM documentation supersedes: N/A

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Approved by:	Eugene Kelly	/	Eugene M. Kelly	/	9/27/17

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 (LAR) (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 and Type C Leak Rate Testing frequencies. By letter dated July 17, 2017 (ADAMS Accession No. ML17198A229), EGC supplemented the LAR with additional information in response to the U.S. Nuclear Regulatory Commission (NRC) staff’s requests (ADAMS Accession Nos. ML17180A153 and ML17198A229). The NRC staff has reviewed the additional information and has determined that further information below is needed to support the NRC’s staff’s continued technical review of the LAR.

RAI 5-A

Section 4.2.6 of EPRI [Electric Power Research Institute] TR [Topical Report]-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 [core damage frequency],” therefore requiring a risk assessment. In response to request for additional information (RAI) 5, EGC described the probabilistic risk assessment (PRA) modeling to estimate the change in risk from loss of NPSH to the ECCS pumps and provided a ΔCDF estimate of 2.4E-8/year.

(a) As described in the RAI response, the risk analysis appears to assume a total loss of containment heat removal for all accident scenarios that were considered, such as transients, loss of coolant accident, anticipated transient without scram, special initiators, etc., which are listed in Table 5-1 of the RAI response. However no justification was provided for scenarios with containment decay heat removal available. Explain and justify why the loss of containment overpressure impacting NPSH for the ECCS injection is not a concern in any accident scenario with containment decay heat removal available. Alternatively, if any additional accident scenarios are identified to contribute to the risk increase, provide an updated estimate of ΔCDF.

(b) The RAI response attempts to explain the PRA modeling for scenarios “post containment failure” and for scenarios with “successful pool venting” in Figures 5-4 and 5-5. Since the containment would already be failed due to the postulated pre-existing containment leak, further PRA modeling appears unnecessary and the application of the model described in Figure 5-1 to those post containment failure scenarios may result in a reduction in the risk estimate. The RAI response also states that for “successful pool venting,” only sources outside the containment are credited. Clarify how the PRA model described in response to RAI 5 correctly captures the risk impact from accident scenarios “post containment failure” and those with “successful pool venting.”

(c) In response to RAI 5.c, ΔLERF (large early release frequency) resulting from loss of NPSH was equated to ΔCDF. If a new method to estimate ΔLERF is deemed necessary in response to items a or b above, describe and justify any credit taken for reducing ΔLERF below the value for ΔCDF.

For ease of review the responses below will be prefaced with the applicable NRC request for additional information portion.

NRC RAI 5-A(a)

As described in the RAI response, the risk analysis appears to assume a total loss of containment heat removal for all accident scenarios that were considered, such as transients, loss of coolant accident, anticipated transient without scram, special initiators, etc., which are listed in Table 5-1 of the RAI response. However no justification was provided for scenarios with containment decay heat removal available. Explain and justify why the loss of containment overpressure impacting NPSH for the ECCS injection is not a concern in any accident scenario with containment decay heat removal available. Alternatively, if any additional accident scenarios are identified to contribute to the risk increase, provide an updated estimate of ΔCDF.

Response to 5-A(a):

When containment decay heat removal is available the reduction of NPSH associated with a pre-existing containment leak as evaluated in the ILRT risk assessment is not sufficient to fail the RHR LPCI or Core Spray (CS) pumps (i.e., sufficient NPSH remains to support continued pump operation). This was confirmed by executing Quad Cities specific MAAP calculations.

Table 5A-1 below presents the results of two MAAP calculations performed to confirm that sufficient NPSH exists to support continued pump operation for the RHR LPCI and CS systems. These MAAP calculations used bounding assumptions and inputs with regards to the ILRT risk assessment methodology, as follows:

- Both MAAP cases assume a design basis large break LOCA (i.e., 28" diameter recirculation line break), which has the greatest potential to increase the temperature of the suppression pool and lead to a loss of NPSH challenge.
- A single train of RHR containment heat removal is operating in Suppression Pool Cooling (SPC) mode.
- Pre-existing containment leakage is assumed to be 200 La (rather than 100 La specified in the ILRT methodology) to demonstrate margin with respect to postulated leakage.
- Both RHR LPCI and CS pumps were evaluated.

These MAAP calculations demonstrate that with a single train of containment heat removal operating the suppression pool temperature remains well below 212^{o}F, a temperature level that can challenge NPSH requirements. The MAAP results show that even for a 200 La containment leakage rate, the suppression pool temperature remains below approximately 174^{o}F and the pumps have sufficient NPSH to remain operational.

These MAAP calculations were reviewed for inputs and key assumptions that could potentially be non-conservative and impact the loss of NPSH assessment.

Key assumptions and inputs examined include:

1. RHR containment heat removal start time – For each MAAP calculation SPC is initiated 10 minutes after the suppression pool temperature level rises above 95°F. This takes into account time for alignment and is consistent with the current Quad Cities EOPs, which require operators to hold suppression pool temperature below 95°F.
2. Initial suppression pool water temperature – The initial water temperature is conservatively set at 95°F, the highest temperature allowed by the EOPs without needing suppression pool cooling.
3. Initial pool water inventory – The initial suppression pool water level is set at a best-estimate height of 14.05 ft which is consistent with typical plant operation.
4. RHRSW temperature - The RHRSW river water temperature is assumed to be 95°F based on the Quad Cities UFSAR and is judged to be a conservative value.
5. Reactor Power level – Reactor power level is set at 100%, the highest rated core power level, for all calculations.
6. Large LOCA MAAP Modeling issues – MAAP is known to have some modeling issues (e.g., potential for reverse flow not modeled) that introduce uncertainty for Large LOCA scenarios. These uncertainties only are applicable to results in the early portion of the run (i.e., approximately first three minutes) prior to core recovery. These deficiencies do not impact the ILRT MAAP calculation results since the peak suppression pool temperature is reached hours into each run, which is well beyond the calculational timeframe during which the uncertainty is greatest.
7. Pump flow rates – Pump flow rates are based on system analysis data for flow capabilities of the respective pumps.
8. MAAP NPSH calculation – MAAP evaluates (and will fail) injection sources based on NPSH requirements. The MAAP NPSH calculation includes control volume pressures and vertical heights (i.e., static head) but ignores flow losses (e.g., pipe friction and line losses). Ignoring pipe flow losses is non-conservative, but these losses are very small contributors in comparison to the factors included in the MAAP NPSH calculation. In the Quad Cities MAAP calculations, the NPSH margin available is large compared to the potential impact of line losses.

The above assumptions and inputs are considered to be either best estimate or conservative, except for the non-conservative modeling aspect associated with pipe flow losses (which is judged a negligible contributor). Based on the above, the MAAP calculations are judged to present a best estimate result for evaluating the potential loss of NPSH. As noted previously, the MAAP case assumptions of postulating a large LOCA and applying a containment leakage of 200 La are judged to provide margin in the MAAP calculation results. It is additionally noted that industry testing and analysis indicates that ECCS pumps used in BWR 3/4 plants are capable of adequate short term (~24 hr) operation well below the manufacturer's recommended design NPSH (e.g., 65% of the specified NPSH limit for Brown Ferry as documented in

NUREG/CR-2973). Therefore, additional margin exists beyond that reflected in the MAAP calculations.

TABLE 5A-1

MAAP RESULTS(1)

LLOCA CASE	CONTAINMENT LEAK SIZE EQUIVALENCE	INJECTION LOST AT TIME	TIME TO CORE DAMAGE	PEAK SUPPRESSION POOL TEMP.	INITIAL CONDITIONS
QC05024-200XLEAK-LPCS	200 La	N/A	N/A	174°F	2 CS pumps, 1 RHR pump in SPC mode
QC05024-200XLEAK-LPCI	200 La	N/A	N/A	174°F	2 RHR pumps in LPCI mode (1 loop), 1 RHR pump in SPC mode(2)

Notes to Table 5A-1:

(1) All MAAP simulation durations are 24 hours.

(2) A LLOCA simulation was also run with 1 RHR pump in LPCI mode and 1 RHR pump in SPC mode. Note, that the Quad Cities RHR system consists of two loops with 2 RHR pump trains in each loop. Peak suppression pool temperature was 173°F and NPSH was also sufficient.

Conclusions

MAAP calculations performed in support of the Quad Cities PRA demonstrate that if RHR containment heat removal is available, the suppression pool water temperature stays well below 212 °F in the long term for a large LOCA and reduction in NPSH for LPCI and CS injection is not a concern (i.e., the LPCI and CS pumps continue to operate). Table 5A-1 presents the results from the two bounding MAAP sensitivity cases performed in support of the ILRT analysis.

NRC RAI 5-A(b)

> The RAI response attempts to explain the PRA modeling for scenarios “post containment failure” and for scenarios with “successful pool venting” in Figures 5-4 and 5-5. Since the containment would already be failed due to the postulated pre-existing containment leak, further PRA modeling appears unnecessary and the application of the model described in Figure 5-1 to those post containment failure scenarios may result in a reduction in the risk estimate. The RAI response also states that for “successful pool venting,” only sources outside the containment are credited. Clarify how the PRA model described in response to RAI 5 correctly captures the risk impact from accident scenarios “post containment failure” and those with “successful pool venting.”

Response to 5-A(b):

In the PRA model, the loss of NPSH fails CS and RHR (from the suppression pool) for RPV injection early in the event tree sequences. This failure is maintained throughout the accident sequences, such as in consideration of RPV injection later in a given sequence. Figures 5-4 and 5-5 were included in the previous RAI response for completeness purposes. These figures demonstrate that upon loss of NPSH due to a pre-existing containment failure, the CS and RHR RPV injection systems taking suction from the suppression pool are failed throughout the model, including in the model logic that assesses the available RPV injection systems later in sequences during consideration of available RPV injection post-containment failure due to containment over-pressure failures. Containment overpressure scenarios and containment vent scenarios are further discussed below.

<u>Containment Overpressure Scenario</u>

In the event that containment heat removal (i.e., Main Condenser, RHR in SPC or SDC, and containment vent) is unavailable, containment pressure will increase to the point of containment failure due to over-pressure. In the Quad Cities PRA, if containment pressure increases to the point of containment failure, then a probability is assigned to fail core injection due to the potential for disruption of injection lines and degraded environmental conditions in plant areas housing injection equipment. This scenario is represented in the PRA model with basic event 1CNPVDWRUPT--R-- as shown in Figure 5-4 of the previous RAI response, also reproduced below. This high pressure containment rupture scenario is different than a pre-existing leak. A pre-existing containment failure (as might be detected by the Type A ILRT) coupled with loss of SPC or SDC is independent of an over-pressure event, however, the end result would be the same, i.e., loss of ECCS injection leading to core damage. Therefore, there is no change in CDF associated with post-containment failure due to containment over-pressure failures.

The ECCS pumps taking suction from the suppression pool are failed in the model prior to containment over-pressure failure and the pumps are not credited for post-containment failure RPV injection. This is reflected in Figure 5-4 under gate ZZ-FAIL-EC which considers RPV injection sources that may be available if not disrupted by a containment over-pressure failure. Under this gate RHR LPCI and CS, when taking suction from the suppression pool, are shown as failed (due to earlier loss of SPC, SDC and NPSH associated with pre-existing containment leakage). It is additionally noted that no credit is applied in the model for the potential that the postulated ILRT pre-existing leakage could mitigate a containment over-pressure condition and prevent a containment rupture and the associated physical disruptions of injection lines.

<u>Containment Venting Scenario</u>

As noted in the previous paragraph, containment venting is a means of removing decay heat from containment. Decay heat removal using the containment vent system will support RPV injection from a water source outside containment (i.e., Feedwater, CRD, FP, SSMP and Condensate). Model logic associated with a venting scenario is shown in Figure 5-5 of the previous RAI response and also reproduced below. Core Spray and LPCI are failed (shown as red basic events) in Figure 5-5 as a result of loss of NPSH caused by loss SPC and SDC, and a pre-existing leak. Figure 5-5 also shows that CS and LPCI realignment to the Clean Condensate Storage Tank (CCST) is not credited in the base model (an application specific model change to credit realignment is shown in Figure 5-6). Venting success has no impact on CS and LPCI injection success as CS and LPCI injection are already failed due to a loss of NPSH caused by a pre-existing leak and failure of containment heat removal. Therefore, there is no change in CDF associated with containment venting in the Quad Cities base model. An application specific PRA model allows credit for containment venting provided operators align CS or RHR suction to the Clean Condensate Storage Tank (CCST).

The following summarizes the risk impact from accident scenarios “post containment failure” and those with “successful pool venting”:

- In the base PRA model there is no contribution to delta CDF or delta LERF associated with loss of NPSH for CS and RHR LPCI in the ILRT calculations associated with the model logic portions for post containment failure or successful pool venting. For these cases, the PRA model already considers the containment to be failed (via the pre-existing leakage), and the CS and RHR LPCI RPV injection to be failed (prior to containment vent or structural containment failure) due to loss of NPSH.
- The application specific model used in the previous RAI response credits CS and RHR LPCI for RPV injection for containment venting scenarios provided operators align a CS or RHR LPCI pump suction to the Clean Condensate Storage Tank (CCST). This is reflected in the ΔCDF and ΔLERF values provided in the July 27, 2017 license amendment request (LAR) RAI response.
- In consideration of structural containment failure due containment over-pressure following loss of containment heat removal, the model does not credit the fact that the pre-existing containment failure evaluated in the ILRT risk assessment may sufficiently relieve the containment pressure increase to preclude a structural containment failure. In this respect, the ILRT risk assessment model is potentially conservative in that if structural failure does not occur due to pressure relief via the pre-existing leak, other injection systems that are not impacted by NPSH requirements from the suppression pool could have an increased probability of remaining available following a less energetic containment failure.

<u>Additional Information</u>

To assist in understanding of the PRA modeling, a short summary of the purpose of the RAI 5 figures is presented (the RAI figures from the previous RAI are presented again below):

1. Figure 5-1 shows the pre-existing failure basic in the FPIE PRA model. This was the event whose failure probability escalated to reflect the ILRT extension.
2. Figure 5-2 shows propagation of the basic event from Figure 5-1 to ECCS injection logic. RHR LPCI and CS injection are shown failed given no Suppression Pool Cooling (SPC) or Shutdown Cooling (SDC).
3. Figure 5-3 shows that when ECCS-NPSH logic is failed as shown in Figure 5-2, it leads to failure of CS and LPCI.
4. In the base PRA model, a pre-existing leak without SPC/SDC results in a reduction in NPSH and failure of ECCS injection. If Operators switch suction for CS or RHR LPCI to the CCST, NPSH challenges associated with the suppression pool would not prevent their operation. Figure 5-6 shows an application specific PRA model change crediting the CS and LPCI make-up source outside of containment.
5. Figure 5-4 and Figure 5-5 were provided for completeness. Figures 5-4 and 5-5 illustrate other portions of the logic impacted by the ECCS-NSPH logic gate.
6. Figure 5-4 shows the logic for RPV injection post-containment failure. As discussed previously, this logic addresses random failure of systems (under gate ZZ-FAIL-EC) and failure from a large DW containment rupture failure that could

cause loss of injection from all sources. A large DW containment failure assumes the potential exists for a catastrophic failure, and this failure would not be associated with a pre-existing leak. Figure 5-4 also shows the impact of a loss of LPCI and CS caused by loss of NPSH (as represented by ECCS-NPSH logic failure). FW, SSMP and CRD random failure logic is not impacted by ECCS-NPSH gate failure.

7. Figure 5-5 shows that systems such as FW, SSMP, CRD, FP and Condensate may be credited in some scenarios as these systems take suction from outside containment and are not impacted by a loss of NPSH caused by a pre-existing leak.
8. Figure 5-6 (also discussed above) shows an application specific PRA model change to credit RHR LPCI and CS injection following alignment to the CCST[(1)].

(1) CCST is shortened to CST in PRA model basic event description.

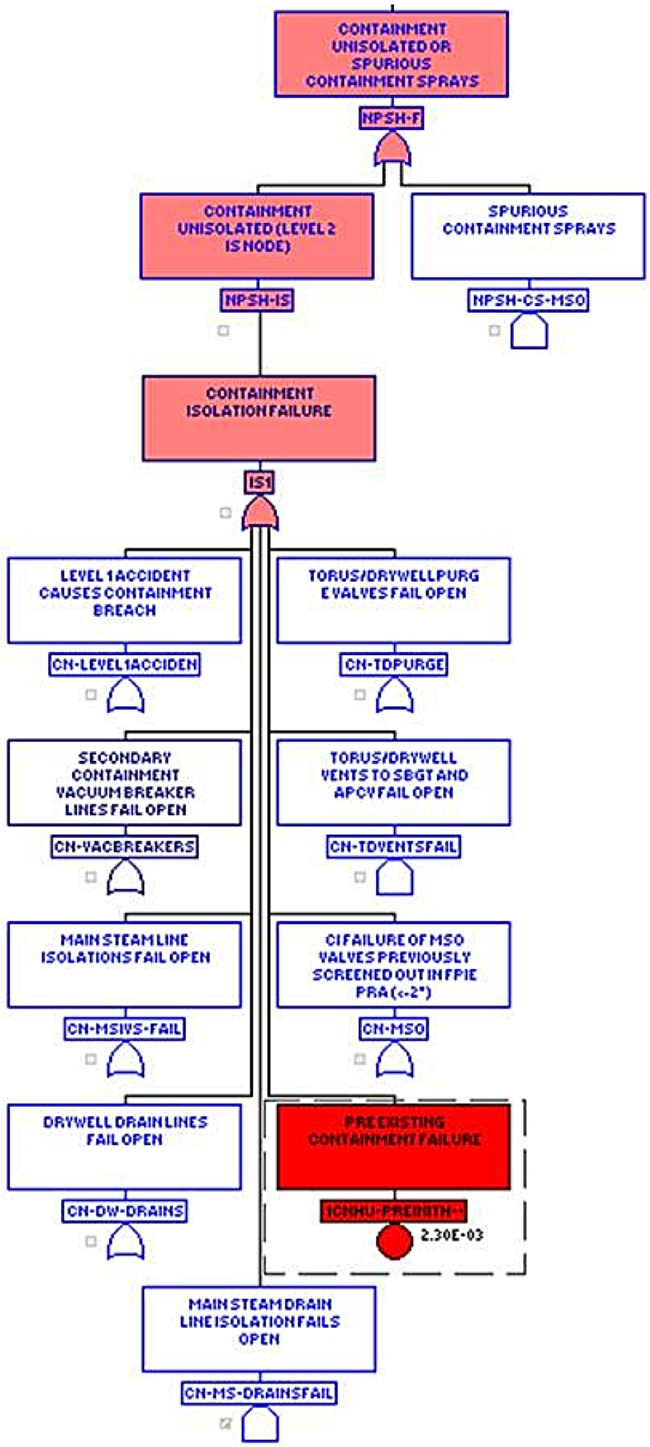

FIGURE 5-1

PRE-EXISTING CONTAINMENT FAILURE MODEL LOGIC

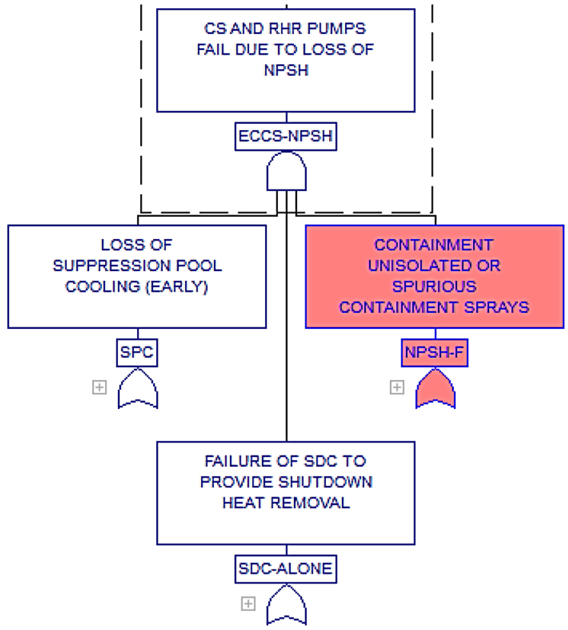

FIGURE 5-2

ECCS-NPSH FAILURE LOGIC

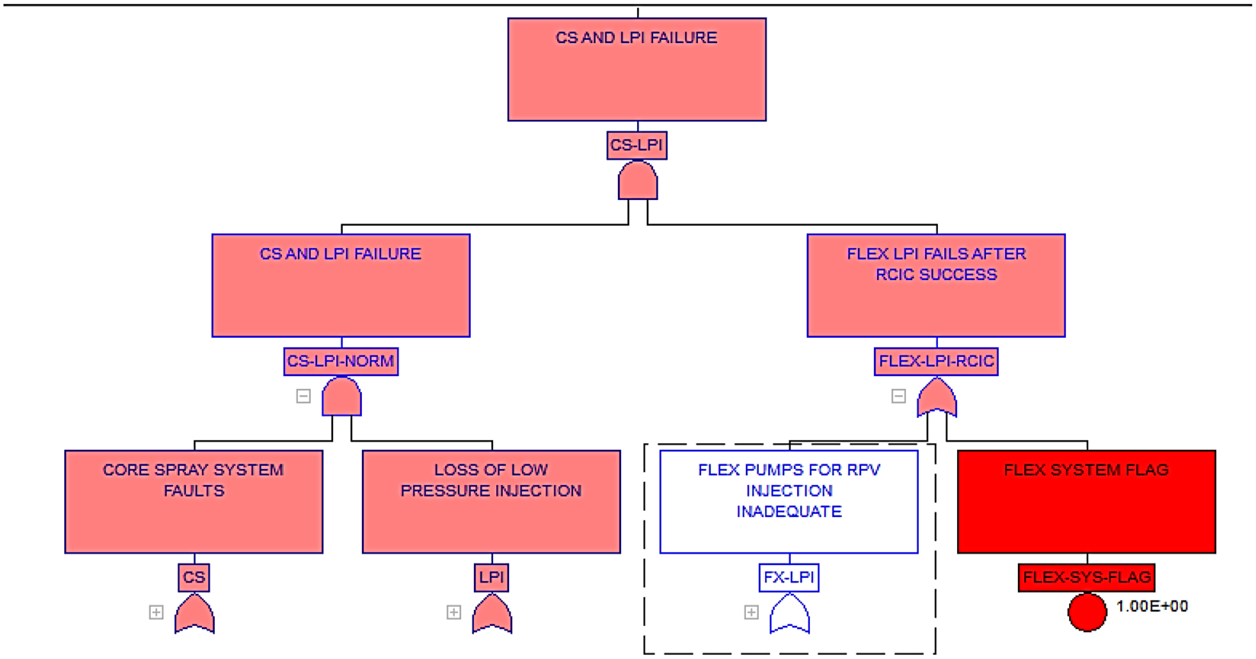

FIGURE 5-3

CS-LPI MODEL LOGIC

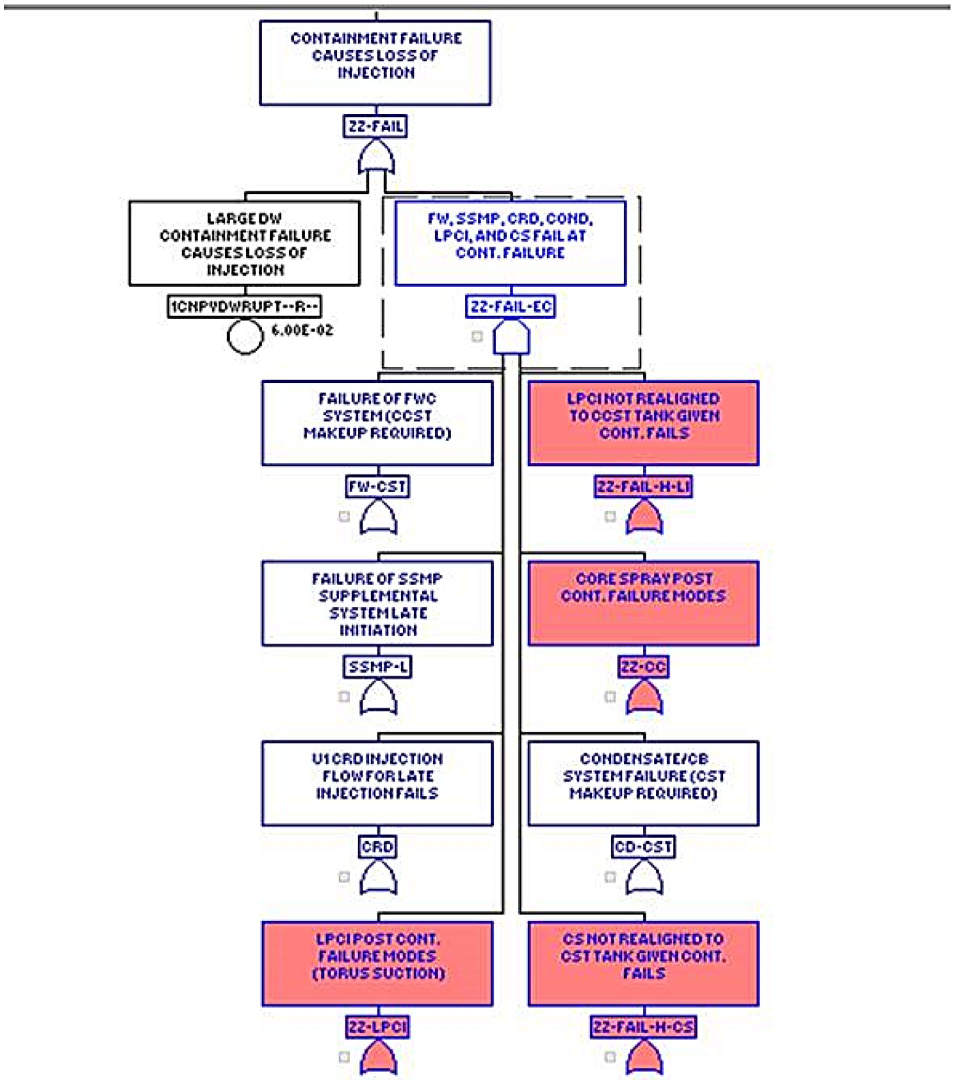

FIGURE 5-4

ECCS-NPSH FAILURE IMPACT TO ZZ-FAIL

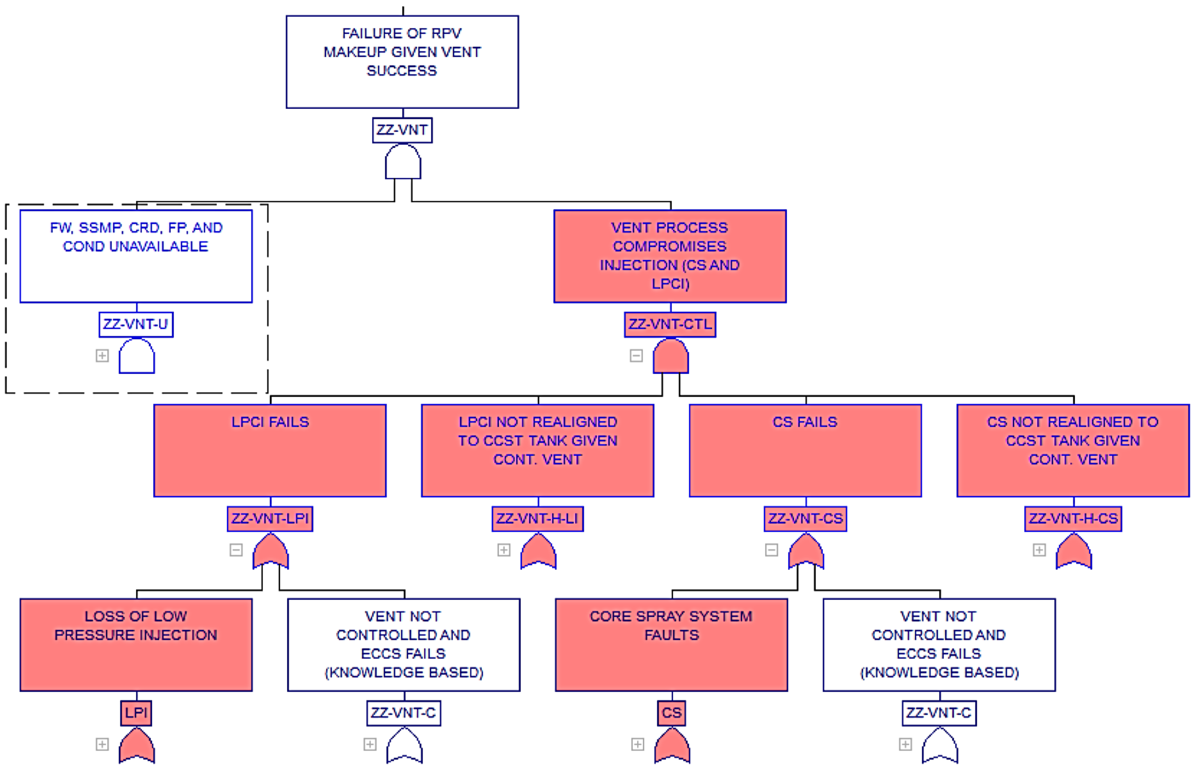

FIGURE 5-5

ECCS-NPSH FAILURE IMPACT TO ZZ-VNT

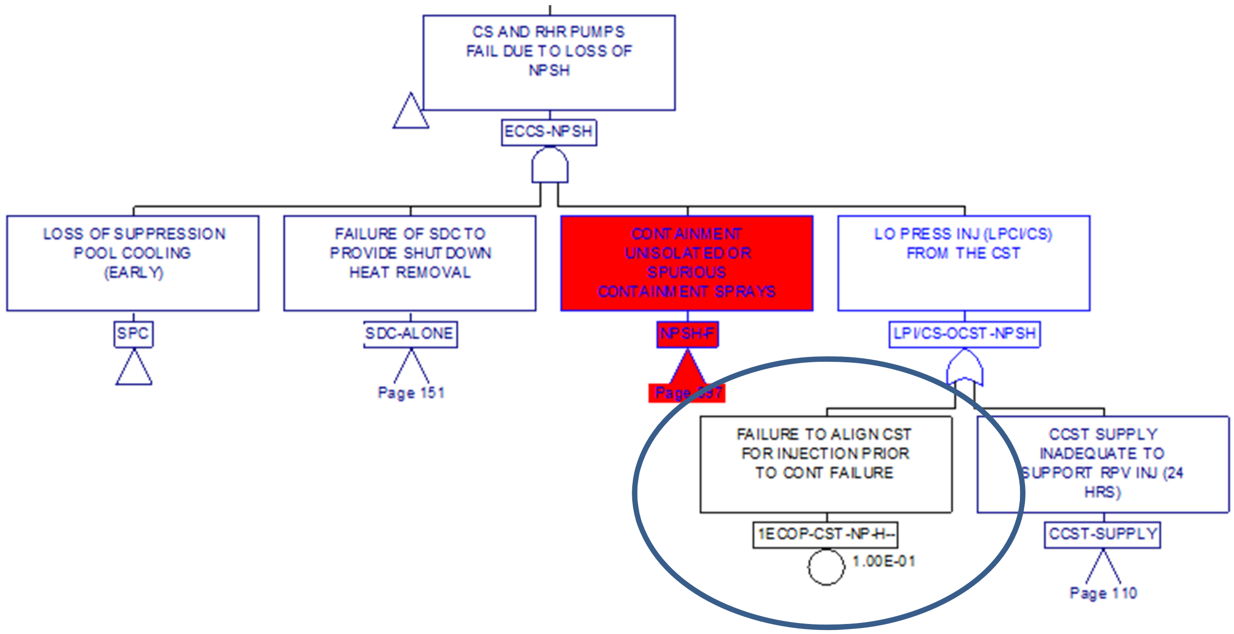

FIGURE 5-6

NEW OPERATOR ACTION LOGIC

NRC RAI 5-A(c)

> In response to RAI 5.c, ΔLERF (large early release frequency) resulting from loss of NPSH was equated to ΔCDF. If a new method to estimate ΔLERF is deemed necessary in response to items a or b above, describe and justify any credit taken for reducing ΔLERF below the value for ΔCDF.

Response to 5-A(c):

Based on the responses to RAI 5-A(a) and 5-A(b) above, there is no need to recalculate ΔCDF or ΔLERF. Since there is no need to recalculate ΔCDF or ΔLERF, there is no need to use a new method to estimate ΔLERF. The conservative assumption that ΔLERF = ΔCDF is acceptable to support the ILRT risk acceptance criteria.