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

Subject: Additional Information Supporting Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program

Reference:

1. Letter from J. L. Hansen (Exelon Generation Company, LLC) to U. S. NRC, "Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)" dated February 16, 2010

2. Letter from C. Gratton (U. S. NRC) to C. G. Pardee (Exelon Generation Company, LLC), "Quad Cities Nuclear Power Station, Units 1 and 2 - Request for Additional Information Regarding an Amendment Request to Relocate Specific Surveillance Frequency Requirements to a Licensee-Controlled Program (TAC Nos. ME3374 thru ME3375)," dated May 14, 2010.

3. May 20, 2010 Teleconference between U. S. NRC (C. Gratton) and Exelon Generation Company, LLC (J. Schrage, et al)

In Reference 1, Exelon Generation Company, LLC (EGC) submitted a request to amend Appendix A, "Technical Specifications," (TS) of Renewed Facility Operating License Nos. DPR-29, and DPR-30 Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively. The proposed amendment revises the QCNPS TS by relocating specific surveillance frequencies to a licensee-controlled program.

In Reference 2, the NRC forwarded six requests for additional information concerning the Reference 1 license amendment request. In the Reference 3 teleconference representatives from the NRC and EGC clarified the information that was requested by the NRC. The attachments to this letter provide the additional information requested by the NRC.

There are no regulatory commitments in this letter or the attachment.

Should you have any questions or require additional information, please contact Mr. John L. Schrage at (630) 657-2821.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 22nd day of June 2010.

Respectfully,

Jeffrey L. Hansen
Manager - Licensing

Attachment 1: Response to NRC Request for Additional Information, License Amendment Request to Relocate Specific Surveillance Frequency Requirements to a Licensee-Controlled Program

Attachment 2: Response to NRC Request for Additional Information, Updated Table 2.2-1, Page 9, Status of Identified Gaps to Capability Category II of the ASME PRA Standard

ATTACHMENT 1
Response to NRC Request for Additional Information

License Amendment Request to Relocate Specific Surveillance Frequency Requirements to a Licensee-Controlled Program

By letter dated February 16, 2010, Exelon Generation Company, LLC (EGC) submitted a request to amend Appendix A, "Technical Specifications," (TS) of Renewed Facility Operating License Nos. DPR-29, and DPR-30 Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, respectively. The proposed amendment revises the QCNPS TS by relocating specific surveillance frequencies to a licensee-controlled program.

The NRC forwarded six requests for additional information concerning the proposed license amendment request by letter dated May 14, 2010. On May 20, 2010, representatives from the NRC and EGC clarified the information that was requested by the NRC. The EGC response to the NRC requests is provided below.

NRC Request for Additional Information (RAI)-01

"Exelon Generation Company letter dated February 16, 2010 (Agencywide Documents Access and Management System Accession No. ML100480339), proposed to relocate specific surveillance frequencies to a licensee-controlled program through the implementation of Nuclear Energy Institute 04-10, "Risk-informed Technical Specifications Initiative 5b, Risk-informed Method for Control of Surveillance Frequencies," Revision 1. The application included a gap analysis (i.e., self-assessment) for the Quad Cities Nuclear Power Station (QCNPS) probabilistic risk assessment (PRA) model was completed in 2004. This gap analysis was performed against the American Society of Mechanical Engineers [ASME] PRA Standard, Addendum A (the Standard). The 2004 gap analysis defined a list of 85 supporting requirements from the Standard for which potential gaps to the Standard were identified. The Nuclear Regulatory Commission (NRC) staff has determined that the following information is needed in order to complete its review:

1. In Table 2.2-1 of Attachment 2 of the submittal, Gap #2 identifies the need for additional investigation to assure appropriate components and failure modes are modeled. Gap #3 identifies that component and failure mode exclusion criteria are not documented. The licensee disposition for these items states that the PRA model is "...judged to include proper treatment of components and failure modes for Capability Category II requirements," and that documentation enhancements are all that is required. Supporting Requirement SY-A12 of the PRA internal events standard identifies that failure modes of components which are beneficial to system operation should not be included, and SY-A12 does not distinguish unique capability categories. Similarly SY-A15 provides quantitative criteria for exclusion of components and failure modes without distinguishing capability categories. The licensee's disposition refers to Capability Category II requirements which do not exist for these supporting requirements, and refer to SY-A14 (Gap #3) when it appears that the deficiency is against SY-A15.

   Therefore, the disposition of these gaps is not understood by the NRC staff, and further clarification is needed. Identify how the QCNPS PRA model addresses failure modes beneficial to system operation, and justify the basis for the judgment that this gap is not significant, or is only a documentation issue and not a technical issue. Similarly discuss

the criteria used in the QCNPS PRA model to exclude components and failure modes, and the basis for the judgment of the gap significance."

EGC Response

- ***"Therefore, the disposition of these gaps is not understood by the NRC staff, and further clarification is needed."***

  During the May 20, 2010 teleconference, EGC clarified to the NRC that the Supporting Requirement (SR) gaps identified in Table 2.2-1 of Attachment 2 to the February 16, 2010 license amendment request (LAR) refer to ASME PRA Standard, Addendum B (ASME RA-Sb-2005) requirements, and not ASME PRA Standard, Addendum A (ASME/ANS RA-Sa-2009) requirements.
  Table 1 below provides a cross reference of the ASME PRA Standard, Addendum A SRs identified by the NRC RAI to the equivalent SR in ASME PRA Standard, Addendum B.

Table 1
Cross Reference of Renumbered Supporting Requirements Impacting QC TSTF-425 RAIs

Summary of Technical Issue for Supporting Requirement (SR)	Associated SR in ASME PRA Standard, Addendum B (ASME RA-Sb-2005)	Associated SR in "Combined PRA Standard" (ASME/ANS RA-Sa-2009)
Include those failure modes that would affect system operability.	SY-A12	SY-A11
Component failure modes may be excluded if the contribution is less than 1% of the total failure rate or probability for that component.	SY-A14	SY-A15
Evaluate the duration of the time that equipment was unavailable.	DA-C12	DA-C13

- ***"Identify how the QCNPS PRA model addresses failure modes beneficial to system operation, and justify the basis for the judgment that this gap is not significant, or is only a documentation issue and not a technical issue."***

  During the May 20, 2010 teleconference, the NRC clarified that the specific request concerning SR SY-A12 (i.e., exclusion of failure modes of components which are beneficial to system operation) refers to SY-A12 in ASME/ANS RA-Sa-2009 (i.e., the Combined Standard). As delineated in Table 1 above, the SR SY-A12 gap in the February 16, 2010

LAR corresponds to SY-A11 in the Combined Standard. EGC did not identify SR SY-A12 from the Combined Standard as a gap in the QCNPS PRA model self-assessment. With respect to SR SY-A12 (Gap #2) in ASME RA-Sb-2005, the QCNPS PRA explicitly models active failure modes such as Failure to Start, Failure to Run, Failure to Open, and Failure to Close. Certain low probability component failure modes (e.g., Failure to Remain Open, Failure to Remain Closed, Plugging, or other passive failure modes) may not always be explicitly modeled if the failure mode represents less than 1% of the total component failure probability as stated in SY-A14 in ASME RA-Sb-2005. Passive failure modes (e.g., Bus Circuit Breaker Fails to Remain Closed) are explicitly modeled, as needed, to model appropriate failure modes and dependencies.

The intent of ASME RA-Sb-2005 SRs SY-A12 and SY-A14 in the QCNPS PRA model is satisfied. The documentation of the screening of these very low probability failure modes does not affect the PRA results. However, EGC has identified this documentation as an improvement item.

The failure modes that would be examined as part of a change to a relocated surveillance frequency (i.e., in a licensee-controlled program) would be active component failures, which are the same failure modes that are included in the PRA. Therefore, the QCNPS PRA model is at the appropriate level of detail for this application.

- ***The licensee's disposition refers to Capability Category II requirements which do not exist for these supporting requirements, and refer to SY-A14 (Gap #3) when it appears that the deficiency is against SY-A15.***

  In Table 2.2-1 of Attachment 2 of the LAR, SY-A12 and SY-A14 (i.e., from ASME RA-Sb-2005) are identified to meet "Capability Category II requirements." SRs SY-A12 and SY-A14 from ASME RA-Sb-2005 do not distinguish between PRA Capability Category I, II, or III. For these SRs, the SR is either "Met (All)" or "Not Met". The intent of the reference to "Capability Category II requirements" is that the QCNPS PRA treatment of these SRs would support PRA applications that require PRA quality meeting "Capability Category II". The discussion for SY-A12 and SY-A14 is not intended to imply that these SRs were specifically assigned "Capability Category II" based on the QCNPS self assessment.

NRC RAI-02

"In Table 2.2-1 of Attachment 2 of the submittal, Gaps # 7, # 8, and # 9 identify that the number of demands and the standby times for components are estimated rather than directly calculated from plant-specific records, and this is "...judged to appropriately estimate ..." these parameters. Justify the basis for the judgment that estimates, rather than actual data, are appropriate."

EGC Response

Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b Risk-Informed Method for Control of Surveillance Frequencies," Step 8, "Associated STI SSC

Modeled in PRA?," requires the determination of an appropriate time-related failure contribution for the applicable components to be analyzed. Additionally, NEI 04-10 states that the time-related failure contribution can be based on recognized data sources or plant-specific data. In the QCNPS PRA model, the failures of equipment (i.e., the numerator of the failure rate estimation) are derived from actual QCNPS operational records. Consistent with ASME RA-Sb-2005 SR DA-C7, the number of plant-specific demands (i.e., the denominator of the failure rate estimation) is based on the number of scheduled surveillance tests, which is also consistent with actual practice at QCNPS.

For standby systems, nearly all of the demands are surveillance tests. Basing the number of demands on the surveillance test frequencies is therefore consistent with actual demand experience, but may not account for plant operational demands. The exclusion of these additional demands in the failure rate estimation is conservative in that the denominator of the failure rate estimation would then be equal to or less than the actual number of demands, resulting in a higher estimated failure rate. This difference will not significantly impact the risk profile. Therefore, the use of QCNPS scheduled surveillance frequencies for the number of plant-specific component demands is an appropriate estimate for the plant-specific demands.

Additionally, NEI 04-10, Step 8, requires that an appropriate time-related failure contribution be utilized in the surveillance frequency change assessment, and Step 14, "Perform Sensitivity Studies," requires the performance of sensitivity studies regarding the choice of that value.

NRC RAI-03

"In Table 2.2-1 of Attachment 2 of the submittal, Gap #10 identifies that supporting requirement DA-C10 is not satisfied. It is not clear exactly what the deficiency is based on the entry in the table. Further, the disposition of this item states that surveillance test procedures are "...judged to address the appropriate failure modes with respect to the estimated number of demands." Provide a more specific summary of the deficiency with regards to supporting requirement DA-C10, and justify the basis for the conclusion that the PRA model is actually using appropriate data."

EGC Response

The deficiency described in Gap #10 of the QCNPS PRA model self assessment identifies that the documentation associated with estimation of failure rates should be enhanced to further support the methodology for using surveillance test procedure frequencies versus the number of actual completed surveillance tests.

As identified in the EGC response to NRC RAI-02, the number of QCNPS plant-specific demands is based on the number of surveillance tests which is also consistent with actual practice. This approach captures the appropriate number of demands performed during each surveillance test (i.e., the value reflects the number of demands within the test procedure if more than one). However, this demand estimate does not account or unplanned surveillances. The exclusion of these additional demands in the failure rate estimation is conservative in that the denominator of the failure rate estimation would then be equal to or less than the actual

number of demands, resulting in a higher estimated failure rate. This difference will not significantly impact the risk profile. Therefore, the use of QCNPS scheduled surveillance frequencies for the number of plant-specific component demands is an appropriate estimate for the plant-specific demands.

NRC RAI-04

"In Table 2.2-1 of Attachment 2 of the submittal, Gap #11 identifies that no interviews with plant staff were conducted to confirm the uncertainty associated with maintenance unavailabilities. Supporting requirement DA-C13 requires such interviews for Capability Category II only when reliable start and finish times are unavailable for significant components. The justification provided for not meeting this requirement discusses the inexperience of plant staff to provide insights on maintenance unavailabilities, and that the data actually used is adequate. Confirm whether interviews are required based on the quality of the data for individual maintenance events, and better justify the actual PRA data without reliance on the experience level of plant staff."

EGC Response

Similar to the issue documented in the EGC response to NRC RAI-01 above, Table 2.2-1 of Attachment 2 to the February 16, 2010 LAR, SR DA-C12 is based on ASME RA-Sb-2005. SR DA-C12 was renumbered to SR DA-C13 in the Combined Standard. This renumbering is described in the EGC response to NRC RAI-01. SR DA-C12 from ASME RA-Sb-2005 identifies the following to meet Capability Category II/III:

> "INTERVIEW the plant maintenance and operations staff to generate estimates of ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basic events."

Table 2.2-1 of Attachment 2 to the February 16, 2010 LAR has been updated with respect to Gap #11. The text for "Description of Gap" has been revised to state that the System Engineers are knowledgeable with respect to the maintenance unavailability times as input to the baseline maintenance unavailability probabilities. The updated Table 2.2-1 page is provided as Attachment 2.

The intent of Gap #11 is to enhance QCNPS PRA documentation to note that the System Engineer input is not explicitly used to support the development of Error Factors for maintenance unavailability probabilities as input to the parametric uncertainty analysis.

The maintenance unavailability data is primarily based on information from high quality sources such as the Maintenance Rule database. Therefore, interviews with System Engineers were not required to support the development of maintenance unavailability times, unless the data was not available from the Maintenance Rule database.

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NRC RAI-05

"In Table 2.2-1 of Attachment 2 of the submittal, Gap #12, Gap #17, and Gap #18 identify that human-induced internal flood data and maintenance alignments are not included in the analysis, but are 'judged to have a minor impact.' Provide the basis for this judgment."

EGC Response

The following evaluation provides justification to support that human-induced internal flood and maintenance alignments would have a minor quantitative impact on the Quad Cities internal flooding analysis.

The frequency of unisolated maintenance-induced floods is assessed using the following inputs:

F_m = Frequency of Invasive Maintenance Event while reactor is at power

P_f^i = Probability that isolation fails initially and flood event is initiated

P_I = Probability that isolation of the flood goes unnoticed and flooding proceeds in the long term to adversely impact multiple systems

These inputs are used to assess the total frequency of a large, unisolated, maintenance-induced flood for both the Reactor Building and Turbine Building, using the following equation:

$$F_T = F_m * P_f^i * P_I$$

- **Reactor Building**

The frequency of unmitigated maintenance-induced floods with reactor at-power are calculated as follows:

F_m^{RB} = 1E-1/yr [Estimate for maintenance alignment that leads to a flooding event.]

$P_f^{I,RB}$ = 5E-3

P_I = 1E-2/yr

$F_T^{RB} = F_m^{RB} * P_f^{I,RB} * P_I$ = 1E-1/yr * 5E-3 * 1E-2 = 5E-6/yr

This is approximately three orders of magnitude lower than the Reactor Building flood scenarios that lead to core damage quantified in the QCNPS PRA model. Additionally, this flood distribution occurs over all of the fluid systems in the Reactor Building. When divided among these systems, the frequency is approximately 1E-7/yr-system. Therefore, the use of frequencies derived from EPRI internal flood data evaluation for the Reactor Building in the QCNPS PRA is appropriate.

- **Turbine Building**

The frequency of unmitigated maintenance induced floods with reactor at-power are calculated as follows:

F_m^{TB} = 1E-1/yr [Estimate for maintenance alignment that leads to a flooding event.]

$P_f^{I,TB}$ = 5E-3

P_I = 0.1

$F_T^{TB} = F_m^{TB} * P_f^{I,TB} * P_I$ = 0.1/yr * 5E-3 * 0.1 = 5.0E-5/yr

This is approximately three orders of magnitude lower in frequency than the frequency of Turbine Building Flooding that lead to core damage as quantified in the PRA model. Therefore, these events are adequately covered in the Quad Cities PRA quantification of Turbine Building flooding induced risk.

- **Summary**

Table 2 below summarizes the frequency of unisolated, maintenance-induced major floods at power with those attributable to pipe or component failures.

Table 2
Frequency of Unisolated, Maintenance-Induced Flood

Location	Frequency of Flood Challenge (per yr)	
	SSC or Pipe Rupture(1)	Unisolated Maintenance Induced Major Flood
Reactor Building	3.3E-3	5E-6
Turbine Building	1.1E-2	5E-5

(1) Values taken from QCNPS internal flooding evaluation.

Based on these quantitative comparisons, the maintenance induced major unisolated floods are relatively minor contributors to the internal flood challenges and are bounded by the internal flood calculations.

NRC RAI-06

"In Table 2.2-1 of Attachment 2 of the submittal, Gap #15 identifies that the effects of internal flooding of jet impingement, humidity, condensation, and temperature are not documented, and have not been addressed for turbine building steamline breaks. The disposition explains that jet impingement is "...judged to have a minor or negligible quantitative impact," and that this gap is a documentation issue only. Provide the basis for why jet impingement is judged to be a minor or negligible issue. Provide a disposition for the other flood effects identified. Provide the basis that the internal flooding analysis properly addresses these effects and that the issue is solely documentation."

EGC Response

The deficiency described in Gap #15 identifies that the QCNPS internal flooding documentation should be updated to identify that pipe whip, jet impingement, and other local damage due to high pressure and temperature pipe breaks in the Turbine Building are evaluated separately as part of the Break Outside Containment (BOC) analysis (e.g., Main Steam line break). The intent of Gap#15, and the associated EGC disposition, is to document that the required information is not available as part of internal flooding documentation. Rather, this information is provided in the BOC documentation.

The principal fluid sources located in the Reactor Building and Turbine Building are low pressure sources (i.e., less than 150 psig). The high pressure and temperature sources outside the containment are evaluated in the QCNPS PRA Initiating Events Notebook to address steam environment, pipe whip, jet impingement, and other local damage.

The High Energy Line Break Analysis addresses the effects of breaks in high-energy lines, such as feedwater, main steam, High Pressure Coolant Injection (HPCI), and Reactor Core Isolation Cooling (RCIC) piping. Breaks in this piping are considered low probability events whose consequences have been assessed in separate evaluations for input into the PRA (i.e., as documented in QCNPS Initiating Events Notebook (QC-PSA-001)).

LOCAs are not explicitly treated in the internal flood analysis. LOCAs have been addressed in the following:

- LOCA Inside Containment
- LOCA – ISLOCA
- LOCA – BOC Breaks Outside Containment. High energy pipe breaks that could disable equipment have been treated in the Break Outside Containment and the ISLOCA evaluation. Therefore, these events are not also considered as flood hazards.

These three categories address those pipe ruptures or component ruptures that release the primary system fluid. Thus, these high-energy lines are already considered in the PRA and therefore, are outside the scope of the internal flood analysis.

ATTACHMENT 2
Response to NRC Request for Additional Information

Updated Table 2.2-1, Page 9
Status of Identified Gaps to Capability Category II of the ASME PRA Standard

Table 2.2-1
Status of Identified Gaps to Capability Category II of the ASME PRA Standard[1]

Title	Description of Gap	Applicable SRs	Current Status / Comment	Importance to Application
Gap #11	It is noted that the System Engineers are knowledgeable with respect to system unavailability times and the potential ranges of system unavailability times as input to the baseline mean system unavailability probabilities. However, no interviews of plant staff were performed to generate uncertainty estimates of unavailability per maintenance act (e.g., develoment of Error Factors to support the parametric uncertainty analyses). Enhance the PRA documentation to note that the System Engineer input is not explicitly used to support the development of Error Factors for maintenance unavailability probabilities as input to the parametric uncertainty analysis.	DA-C12	Open. This deviation from the SR is not considered to significantly alter the PRA qualitative or quantitative results becase it only influences the determination of the error factors assigned to maintenance unavailabilities.	Not significant because the model is consistent with data from the plant MR database, so there will not be an impact on unavailability hours used in the model.

(1) This table is an excerpt from Table 2.2-1 of the LAR with text modified to respond to NRC RAI-04.