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RS-10-142

August 19, 2010

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

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

References:

1. Letter from P. R. Simpson (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 15, 2010

2. Letter from M. J. David (U.S. NRC) to M. J. Pacilio (Exelon Nuclear), "Braidwood Station, Units 1 and 2 – Request for Additional Information Related to License Amendment Request to Adopt TSTF-425, Revision 3 (TAC Nos. ME3370 and ME3371)," dated July 20, 2010

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2, respectively. The proposed change modifies the Braidwood Station Technical Specifications (TS) by relocating specific surveillance frequencies to a licensee-controlled program. The NRC requested additional information to support 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 Attachments 6 and 1, respectively, 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 19th day of August 2010.

Respectfully,

Jeffrey L. Hansen
Manager – Licensing

Attachment: Response to Request for Additional Information

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

NRC Request 1

In LAR Attachment 1, page 3, item number 3, the licensee made the following statement regarding a variation from TSTF-425:

> The insert provided in TSTF-425 to replace text describing the basis for each Frequency relocated to the Surveillance Frequency Control Program has been revised from, "The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program," to read, "The Frequency may be based on factors such as operating experience, equipment reliability, or plant risk, and is controlled under the Surveillance Frequency Control Program." This deviation is necessary to reflect the Braidwood Station basis for frequencies that do not, in all cases, base Frequency on operating experience, equipment reliability, and plant risk.

While the above TSTF-425 deviation from the TSTF-425 TS Bases statement addresses Surveillance Frequencies relocated to, but not changed under, the Surveillance Frequency Control Program (SFCP), it does not specifically exclude Surveillance Frequencies changes made in accordance with the SFCP and is, therefore, not consistent with SFCP requirements. Please provide additional clarification explaining how Braidwood Station intends to ensure that all Surveillance Frequencies relocated to the SFCP, with or without subsequent Frequency changes, will maintain: 1) Bases for unchanged Surveillance Frequencies and, 2) compliance with proposed Braidwood Station TS 5.5.19, "Surveillance Frequency Control Program" requirements.

Response

The proposed change described in Reference 1 requests NRC approval to relocate Surveillance Frequencies to the SFCP. Upon implementation of the proposed change, the existing Technical Specifications (TS) Bases information describing the basis for the Surveillance Frequency will be relocated to the licensee-controlled SFCP. This will ensure that the information describing the bases for unchanged Surveillance Frequencies is maintained.

As discussed in Reference 1, Exelon Generation Company, LLC (EGC) proposed a variation from TSTF-425 that replaced text describing the basis for each Frequency relocated to the SFCP. This variation was necessary because, independent of whether Surveillance Frequencies have been changed under the SFCP, the Surveillance Frequencies are not, in all cases, based on operating experience, equipment reliability, and plant risk.

As required by proposed TS Section 5.5.19, "Surveillance Frequency Control Program," subsequent changes to the Frequencies listed in the SFCP will be made in accordance with the NRC-endorsed methodology described in Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1 (i.e., Reference 3). NEI 04-10 provides the methodology to identify, assess, implement, and monitor proposed changes to Surveillance Frequencies. NEI 04-10 identifies the need to address both quantitative and qualitative considerations when changing Surveillance Frequencies. As discussed in Section 4.0, Step 7, qualitative considerations include vendor-specified maintenance frequency, test intervals specified in applicable industry codes and standards, impact on defense-in-depth protection, and the existence of alternate testing of structures, systems, and components (SSCs) affected by the change. These qualitative considerations provide examples of instances where Surveillance Frequencies changed under the SFCP may not be based upon operating experience, equipment reliability, or plant risk.

As a result, EGC's proposed variation from TSTF-425 provides wording that more accurately reflects the methodology described in NEI 04-10. However, in order to avoid future confusion regarding this issue, EGC will replace the Bases text insert proposed in Reference 1 (i.e., "The Frequency may be based on factors such as operating experience, equipment reliability, or plant risk, and is controlled under the Surveillance Frequency Control Program") with a revised insert that reads "The Surveillance Frequency is controlled under the Surveillance Frequency Control Program." This change to the Bases text insert will be made upon implementation of the proposed change.

NRC Request 2

In Table 2-1 of Attachment 2 of the LAR, Gap #4, regarding the Large Early Release Frequency (LERF) analysis, identifies 19 specific supporting requirement deficiencies to Capability Category II of the American Society of Mechanical Engineers (ASME) probabilistic risk assessment (PRA) Standard, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications." The description of the gap states that the analysis is based on the NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," methodology, and it represents a generally conservative, simplified approach. The importance to the application states that given the conservative nature of the methodology used, the LERF results are also believed to be conservative relative to this application. Supporting requirements LE-D1b and LE-F1a are not necessarily conservative for higher capability categories. Provide an assessment of the specific impact of the deficiency in regards to the application or provide a more thorough basis for why these two supporting requirements are considered more conservative for Capability Category I than II.

Response

Supporting Requirement (SR) LE-D1b provides the following descriptions for Capability Categories I and II:

Capability Category I	Capability Category II
EVALUATE the impact of accident progression conditions on containment seals, penetrations, hatches, drywell heads (BWRs) and vent piping bellows. INCLUDE these impacts as potential containment challenges, as required. An acceptable alternative is the approach in NUREG/CR-6595 [Note (1)].	EVALUATE the impact of accident progression conditions on containment seals, penetrations, hatches, drywell heads (BWRs), and vent pipe bellows. INCLUDE these impacts as potential containment challenges, as required. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.

SR LE-D1b was assessed as meeting Capability Category I with the justification of "The NUREG/CR-6595 approach has been used."

Though the Braidwood Level 2 analysis generally follows the NUREG/CR-6595 approach, the effects of accident progression on containment penetrations (i.e., including seals, hatches, and other possible containment failure locations) have been evaluated as part of the supporting containment analysis originally performed as part of the Braidwood Individual Plant Evaluation (IPE) (i.e., Reference 2). The containment failure characterization is based on structural assessments from the Zion containment structure (i.e., a similar large dry containment), but were supported by a review of the Updated Final Safety Analysis Report (UFSAR) and additional analyses to justify their applicability to Braidwood (IPE Section 4.3.3.1). Containment failure locations were considered during development of the Braidwood applicable containment fragility curves and their effects are therefore included in the analysis of containment performance during accident progression conditions. Therefore, the existing analysis generally meets the intent of LE-D1b Category II.

SR LE-F1a provides the following descriptions for Capability Categories I and II:

Capability Category I	Capability Category II (and III)
IDENTIFY the significant contributors to large early releases (e.g., plant damage states, containment failure modes).	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 4.5.9-3.

The self-assessment evaluation of SR LE-F1a identified that although an assessment by Plant Damage State (PDS) is not currently provided, the information is available to do so. Since the SR wording for Category I indicates "e.g., PDS" but the wording for Category II/III does not include the "e.g.", the Category assignment for this SR is Category I, even though more than an identification of significant contributors has been performed. A review of the sequence data from the current PRA model of record as binned to capture the LERF contributors and PDSs confirms the relative contributions are reasonable for the containment type at Braidwood. In the development of these significant contributors, the simplified LERF model based on NUREG/CR-6595 was utilized.

The process outlined in NEI 04-10, Revision 1 (i.e., Reference 3) for the performance of the evaluation of a change to a surveillance frequency interval requires the development of various sensitivity studies as described below from Step 14.

> Additional sensitivity cases should also be explored for particular areas of uncertainty associated with any of the significant contributors to the CDF and LERF results or if there are open Gap Analysis items when compared to the ASME Standard Capability Category II that would impact the results of the assessment.

The requirement to perform these sensitivity studies as part of a change to a surveillance test interval will ensure that the impact of a gap to the ASME standard is evaluated regardless of whether the difference between capability categories is viewed as conservative or non-conservative.

NRC Request 3

In Table 2-1 of Attachment 2 of the LAR, Gap #7, regarding model uncertainties and related model assumptions, identifies 20 supporting requirement deficiencies to Capability Category II of the ASME PRA Standard. The importance to the application states that each supporting requirement would be addressed by sensitivities per Nuclear Energy Institute (NEI) 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," if applicable to the specific surveillance test interval evaluation. Supporting requirement LE-E4 requires quantification of LERF consistent with the applicable requirements of Tables 4.5.8-2(a), 4.5.8-2(b), and 4.5.8-2(c) of the ASME PRA Standard. The provided description of the gap, current status/comment, and importance to the application are not applicable to this supporting requirement. No adequate basis is provided for the justification of not meeting this supporting requirement; therefore, there is insufficient information for the NRC staff to reach a conclusion on the disposition of this peer review item. Provide a justification and a basis for the impact of not meeting this supporting requirement.

Response

A review of the self-assessment performed to identify gaps between the existing PRA model capability and Capability Category II of the ASME standard has identified an editorial error in the self-assessment document. SR LE-E4 was evaluated as being met (Capability Categories I, II, and III are equivalent for this SR). Therefore, SR LE-E4 should be considered removed from Table 2-1, Gap #7 in Reference 1. This issue has been captured in EGC's Corrective Action Program.

NRC Request 4

In Table 2-1 of Attachment 2 of the LAR, Gap #8, regarding model documentation, identifies 18 supporting requirement deficiencies to the model. The licensee states the importance to the application is that documentation issues do not affect the technical adequacy of the model. The following questions relate to Gap #8:

a. The NRC staff requires clarification for how an internal gap assessment to Capability Category II can be characterized as model documentation issues for technical requirements. The following supporting requirements are not described in the standard as documentation requirements and the NRC staff requests confirmation that these SRs technically meet capability category II: DA-C6; DA-C10; DA-D4; IE-C10; IF-C2; IF-C2c; IF-D7; IF-E3a; SY-A4. Given the importance of surveillance test and component demand data to this application, provide additional clarification for how the licensee plans to meet steps 7 and 8 of NEI 04-10 if DA-C6 and DA-C10 are not properly documented.

b. Of the 64 gaps identified in Attachment 2 Table 2-1, 23 are document-related. The licensee should address how it plans to maintain the long-term configuration management program without adequate documentation. Provide a general timeline for the closure of document-related gaps for the Braidwood PRA.

Response

a. The following information is provided to address supporting requirements DA-C6 and DA-C10.

   Demand data is provided for those plant specific failure mechanisms as listed in the PRA model data notebook. However, the information contained in this notebook does not include the level of detail being requested under either of these supporting requirements to document the process for data collection or the process for review of surveillance tests to confirm the determination of plant specific demands.

   Step 7 of NEI 04-10 (i.e., Reference 3) is related to the identification of non-PRA related qualitative considerations such as surveillance test and performance history, defense-in-depth impact, and other attributes to ensure that the current performance of the SSCs warrants the extension of the surveillance test interval. As Step 7 is more oriented to the identification of qualitative considerations to be addressed by the evaluation, the impact of the status of the documentation associated with demands and surveillance test data would be negligible. Step 8 of Reference 3 identifies actions to take in evaluating the risk characterization of the surveillance test interval. The risk characterization attributes include demand-related contributions to the failure probability. The gap identified for SRs DA-C6 and DA-C10 is addressed through the process outlined in Reference 3 for the development of various sensitivity studies as described below from Step 14.

   > Additional sensitivity cases should also be explored for particular areas of uncertainty associated with any of the significant contributors to the CDF and LERF results or if there are open Gap Analysis items when compared to the ASME Standard Capability Category II that would impact the results of the assessment.

   The requirement to perform these sensitivity studies as part of a change to the surveillance test interval will ensure that the impact of a gap to the ASME standard is evaluated.

   In addition, a specific sensitivity required by Step 14 of Reference 3 is to adjust the failure rate to a factor of three larger, including common cause impacts. As the demand data could potentially impact the failure rate used in the PRA model, this specific sensitivity will provide additional evaluation of the gap associated with SRs DA-C6 and DA-C10.

The remaining supporting requirements are addressed in the table below.

Supporting Requirement	Response
DA-D4	In the development of the data update for the current model of record, a Bayesian update process was utilized and the basis of the applicability of the process is provided in the data notebook. However, for Capability Category II, this SR requires the following: "CHECK that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data." The data notebook did not include a discussion of this reasonableness check at the time of the gap self-assessment. This SR was evaluated as "Not Met" based on a lack of explicit documentation of this reasonableness check even though the application of the Bayesian process was used in the development of the failure rate data and is described in the data notebook. Subsequent to submittal of Reference 1, a description of the check for the reasonableness of this data was added to the data notebook, which closes this gap.
IE-C10	Detailed information related to the development of the plant-specific initiating events is provided in the PRA model initiating events notebook. However, a comparison of the plant-specific values developed and "generic data sources to provide a reasonableness check of the results" was not provided in the notebook. A comparison of the plant specific values and generic industry values for initiating events was performed. This review confirmed the reasonableness of the plant specific data. Subsequent to submittal of Reference 1, the description of this check was added to the initiating event notebook, which closes this gap.
IF-C2	The internal flooding analysis performed for Braidwood includes the compilation of walkdown information. However, the location of flood alarms and flood detection mechanisms were not included in the information collected for this analysis. Based on the specific direction to capture this information in the SR guidance, this SR was considered as "Not Met" even though the walkdown activity appropriately captured the majority of the plant-specific information requested by this SR that is needed to support the internal flooding analysis.
IF-C2c	The current internal flooding analysis does not explicitly address the effects of spray to the level of detail described in the SR guidance. The remainder of the information requested under this SR was identified and evaluated as part of the internal flooding analysis.

Supporting Requirement	Response
IF-D7	The internal flooding analysis notebook does not provide a sufficient level of detail to address the SR guidance related to screening of flood scenario groups. The process and results of screening activities for the internal flooding analysis is provided in the internal flooding notebook. However, as stated above, additional information is required to be included in the notebook to meet the requirements of this SR.
IF-E3a	The internal flooding analysis notebook does not provide a sufficient level of detail to address the SR guidance related to screening of flood areas by the use of conditional core damage probability. The process and results of screening activities for the internal flooding analysis are provided in the internal flooding notebook. However, as stated above, additional information is required to be included in the notebook to meet the requirements of this SR.
SY-A4	The system notebooks developed for the PRA model do not provide evidence of the confirmation of interviews with System Engineers or Operations personnel as required by the guidance contained in this SR. As system notebooks are revised, a review by the system engineer(s) is requested and documented in the notebook.

Similar to SRs DA-C6 and DA-C10, the identified gaps in the above table are also addressed through the process outlined in Reference 3 for the development of various sensitivity studies as described below from Step 14.

> Additional sensitivity cases should also be explored for particular areas of uncertainty associated with any of the significant contributors to the CDF and LERF results or if there are open Gap Analysis items when compared to the ASME Standard Capability Category II that would impact the results of the assessment.

The requirement to perform these sensitivity studies as part of a change to the surveillance test interval will ensure that the impact of a gap to the ASME standard is evaluated.

b. The existence of gaps related to the content of documentation has been addressed in a recent revision to the EGC procedure that governs the development and update of PRA models. The procedure now requires that, prior to the final approval of a model revision, all impacted notebooks must be complete and approved. In addition, the EGC process for performing risk assessments of surveillance interval changes requires gaps to be evaluated for their impact on the proposed change. Therefore, at the time that a surveillance interval change is evaluated, the documentation gaps would either have been closed or will be evaluated as part of the change evaluation in accordance with the SFCP.

References

1. Letter from P. R. Simpson (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 15, 2010

2. Braidwood Individual Plant Examination Report, June 1994

3. Nuclear Energy Institute (NEI) 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," dated April 2007