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

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

Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Response to Request for Additional Information Regarding Application to Revise Braidwood Station and Byron Station Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"

References:

1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Application to Revise Braidwood Station and Byron Station Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b'," dated December 13, 2018 (ADAMS Accession No. ML18352B063)

2) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Supplement to Application to Revise Braidwood Station and Byron Station Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b'," dated February 14, 2019 (ADAMS Accession No. ML19050A399)

3) Email from J. Wiebe (U.S. Nuclear Regulatory Commission) to L. A. Simpson (Exelon Generation Company, LLC), "Partial Issuance of Final RAIs for Braidwood/Byron TSTF-505 Application," dated August 7, 2019 (ADAMS Accession No. ML19232A224)

In EGC letter dated December 13, 2018, as supplemented on February 14, 2019 (References 1 and 2), Exelon Generation Company, LLC (EGC) requested an amendment to the Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2 (BWD), and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2 (BYR), respectively.

The proposed amendments would modify Technical Specifications (TS) requirements to permit the use of risk-informed completion times (RICTs) in accordance with the Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b" (ADAMS Accession No. ML18183A493).

On June 19 and 20, 2019, the U.S. Nuclear Regulatory Commission (NRC) conducted an audit at EGC's offices in Kennett Square, Pennsylvania to support development of its safety evaluation. Upon completion of the audit, the NRC determined that additional information is needed to complete its review of References 1 and 2. The formal request for additional information (RAI) was issued by email on August 7, 2019 (Reference 3).

As documented in Reference 3, the U.S. Nuclear Regulatory Commission has determined that additional information is needed to complete its review of References 1 and 2. Attachment 1 to this letter is a partial response to the requested information. As stated in Reference 3, the response to APLA RAI 02 and APLA RAI 03 is required within 45 days of receipt of Reference 3. These responses will be provided in a subsequent EGC response.

Attachment 2 provides an update to the BYR TS markup pages impacted by these RAI responses. As discussed with Mr. Joel Wiebe on August 22, 2019, EGC will be completing a more detailed review of the TS pages associated with References 1 and 2; if additional TS changes are identified that require changes, a supplement will be provided.

EGC has reviewed the information supporting the No Significant Hazards Consideration and the Environmental Consideration that was previously provided to the NRC in Attachment 1 of the Reference 1 letter. The additional information provided in this submittal does not affect the conclusion that the proposed license amendment does not involve a significant hazards consideration. This additional information also does not affect the conclusion that neither an environmental assessment need be prepared in support of the proposed amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is providing a copy of this letter and its attachment to the State of Illinois.

This letter contains no regulatory commitments. Should you have any questions concerning this submittal, please contact Ms. Lisa Simpson at (630) 657-2815.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 5th day of September 2019.

Respectfully,

Patrick R. Simpson
Sr. Manager – Licensing
Exelon Generation Company, LLC

Attachments:
1) Response to Request for Additional Information
2) Proposed Technical Specifications Changes for Byron Station, Units 1 and 2

cc: NRC Regional Administrator – Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
NRC Project Manager, NRR – Braidwood and Byron Stations
Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Response to Request for Additional Information

By letter dated December 13, 2018, as supplemented on February 14, 2019 (References 1 and 2), Exelon Generation Company, LLC (EGC) requested an amendment to the Renewed Facility Operating License Nos. NPF-72 and NPF-77 for Braidwood Station, Units 1 and 2 (BWD), and Renewed Facility Operating License Nos. NPF-37 and NPF-66 for Byron Station, Units 1 and 2 (BYR), respectively.

The proposed amendments would modify Technical Specifications (TS) requirements to permit the use of risk-informed completion times (RICTs) in accordance with the Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk-Informed TSTF] Initiative 4b" (ADAMS Accession No. ML18183A493).

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As documented in Reference 3, the U.S. Nuclear Regulatory Commission has determined that additional information is needed to complete its review of References 1 and 2. This attachment provides the requested information with the exception of responses to APLA RAI 02 and APLA RAI 03, which will be provided in a subsequent EGC response.

References:

1) Letter from D. M. Gullott (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Application to Revise Braidwood Station and Byron Station Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b'," dated December 13, 2018 (ADAMS Accession No. ML18352B063)

2) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Supplement to Application to Revise Braidwood Station and Byron Station Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b'," dated February 14, 2019 (ADAMS Accession No. ML19050A399)

3) Email from J. Wiebe (U.S. Nuclear Regulatory Commission) to L. A. Simpson (Exelon Generation Company, LLC), "Partial Issuance of Final RAIs for Braidwood/Byron TSTF-505 Application," dated August 7, 2019 (ADAMS Accession No. ML19232A224)

APLA RAI 01 - PRA Facts and Observations (F&Os)

Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities," Revision 2 (ADAMS Accession No. ML090410014), provides guidance for addressing PRA acceptability. RG 1.200 describes a peer review process utilizing the ASME/ANS PRA standard (currently ASME/ANS-RA-Sa-2009, "Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications"[1]) as one acceptable approach for determining the technical adequacy of the

[1] Available from the International Organization for Standardization at https://www.iso.org/store.html.

PRA, once acceptable consensus approaches or models have been established for evaluations that could influence the regulatory decision. The primary result of a peer review are the findings and observations (F&Os) recorded by the peer review and the subsequent resolution of these F&Os. A process to close-out Finding-level F&Os is documented in Appendix X to Nuclear Energy Institute (NEI) 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations," (ADAMS Accession No. ML17086A431) dated February 21, 2017 accepted by the NRC in the letter from Joseph Giitter and Mary Jane Ross-Lee, NRC to Greg Krueger, NEI, dated May 3, 2017 (ADAMS Accession Number ML17079A427).

In Exelon's letter dated December 13, 2018, Tables E2-1 and E2-2 provides the F&Os that remain open after an F&O closure review performed in 2015 for the Byron and Braidwood Stations fire PRAs, along with dispositions of those F&Os for this application. The NRC staff reviewed the dispositions to these F&Os and noted that several F&Os did not include the updates to the F&O resolutions that were documented in Exelon's letter dated June 13, 2018 (ADAMS Accession No. ML18165A181). In its letter dated June 13, 2018, Exelon committed to update the fire PRAs to resolve three of these F&Os prior to implementation of the 10 CFR 50.69 categorization process and to perform a sensitivity study for one of these F&Os as part of the categorization process. In its letter dated December 13, 2018, Exelon identified in Attachment 5 the items that are required to be completed prior to implementation of the Risk Informed Completion Time (RICT) Program at Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2. In its letter dated December 13, 2018, Exelon further states that all issues identified in Attachment 5 will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the RICT Program. Considering these observations address the following:

a. Concerning internal events F&O 1035 and fire F&Os 16-4, 25-11, and 26-9, Exelon stated it would update the PRA model to incorporate the F&O resolutions prior to implementation of the 10 CFR 50.69 categorization process.

    Provide the status of the implementation items and confirm that the proposed resolutions from the 50.69 implementation items remain consistent with the proposed disposition/resolution in the TSTF-505 submittal.

b. The resolution to F&O 20-8 states that three "scaling factors" are used to credit alternate shutdown given abandonment of the main control room (MCR) upon loss of habitability to account for degrees of fire-induced damage. The resolution refers to NRC safety evaluations (SEs) issuing the NFPA 805 amendments for Turkey Point (ADAMS ML15061A237), St. Lucie (ADAMS ML15344A346) and Farley (ADAMS ML14308A048) power plants. The SEs and associated documents refer to three "bounding values" to be assigned to abandonment scenarios depending on the complexity of the abandonment scenarios. The resolution to F&O 20-8 clarifies that an adjustment factor is applied as a multiplier to the control room abandonment cut sets. The SEs and associated documents do not clarify how the bounding scenario values are applied at the cut set level and does not explain how these scaling factors remain bounding when systems, structures, and components are taken out of service during a RICT application.

    i. Describe and justify how the evaluation is developed and applied and how it can be maintained when the plant or the PRA models change and during the translation to the Real-Time risk (RTR) Model.

ii. Provide descriptions of the methodology that was incorporated into the Byron and Braidwood PRAs.

iii. Explain how the bounding scaling factors remain valid when SSCs are taken out of service during a RICT.

EGC RESPONSE to APLA RAI 01 Part a:

(FPIE) F&O 1035:

The resolution of this F&O has been incorporated into the current BYR and BWD internal events model. The PRA model used for RICT will include this resolution.

(Fire) F&O 1035:

Since this F&O is associated with HELB initiators, it has no impact on the Fire PRA quantification. However, the Fire PRA model used for RICT will include this resolution because the Fire PRA uses the FPIE model that addresses this issue as its basis.

(Fire) F&O 16-4:

The review for breaker coordination and the incorporation of load cable failures which result in loss of the associated power supply has been addressed in the Fire PRA model that will be used for RICT.

(Fire) F&O 25-11:

The Fire PRA model that will be used for RICT addresses this F&O, reflecting the use of the latest internal events/flooding PRA model as its basis. The sump clogging value in the Fire PRA reflects the updated sump clogging data used in the internal events/internal flood PRA model that will be used for RICT. The impact of this change was minimal.

(Fire) F&O 26-9:

The Fire PRA that will be used for RICT addresses this F&O and the incorporation of the results of the walkdown of wall mounted panels at both BYR and BWD. The impact of this change was minimal.

EGC RESPONSE to APLA RAI 01 Part b:

i. The methodology used for adjusting the control room abandonment scenario CCDP/CLERP values is described in License Amendment Request (LAR) Table E2-2, Impact to Implementation of RICT, discussion for F&O 20-8. The CCDP and CLERP adjustments are incorporated into the Fire PRA results by adjusting the scenario frequencies to account for the adjusted CCDP and CLERP values for the abandonment cases in the base model quantification.

   To simplify the quantification of the impact of the control room abandonment fire scenarios, a RICT penalty, similar to that applied to BYR for tornado missiles and to both sites for the seismic hazard analysis, will be applied for the control room abandonment

scenarios. If combinations of instruments and controls available only at the remote shutdown panel (RSP) are unavailable based on equipment associated with the configuration applicable to the RICT assessment, the control room abandonment penalty will be included in the RICT calculation. This penalty is calculated as the difference in risk associated with the unadjusted Fire PRA quantification for the control room abandonment scenarios and the risk associated with setting the CCDP contribution for these scenarios to 1.0. Using the unadjusted Fire PRA result provides a conservatively lower baseline risk for calculation of the delta risk associated with the RICT configuration. If the minimum equipment required at the RSP is not available a fire causing control room abandonment will result in core damage. The RTR tool will identify the equipment associated with the instrumentation and controls at the RSP, including required support systems, and will incorporate this penalty as part of the risk increase associated with configurations that make the RSP unavailable. The LERF value used for this penalty will be based on the CLERP to CCDP ratio from the unadjusted control room abandonment scenario quantifications. An alternate adjustment, associated with setting the CLERP value to 1.0, is applied if any containment isolation valves will be unable to isolate given the RICT configuration. This approach is performed using the zero-maintenance model.

The Fire PRA model used for the RICT program will include the update of the penalty values discussed above.

The incorporation of a directly quantified main control room abandonment scenario, using the guidance of NUREG-1921 (Reference RAI 02-1), will be accomplished in the future to provide a more realistic and directly quantifiable FPRA impact on RICT analyses.

ii. See discussion above in item i above regarding the methodology used.

iii. The bounding scaling factors are used to define the baseline risk with a "penalty" that is defined to reflect the impact on control room abandonment of the unavailability of equipment controlled at the remote shutdown panel (RSP). See detailed discussion in item i above.

For control room abandonment the equipment relied upon is that equipment available at the remote shutdown panel and its associated support systems. The impact of this equipment out of service is bounded by the risk penalty values developed through the methodology in item i because these are calculated assuming a CCDP of 1 for the associated scenarios. Other equipment availability will not impact the control room abandonment scenario.

APLA RAI 02 - Key Assumptions and Sources of Uncertainty

As described in Reference 3, EGC will provide the Response to APLA RAI 02 in a subsequent submittal.

APLA RAI 03 - Specific Key Assumptions and Sources of Uncertainty

As described in Reference 3, EGC will provide the Response to APLA RAI 03 in a subsequent submittal.

APLA RAI 04 - Potential Credit for FLEX Equipment or Actions

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ADAMS Accession No. ML17031A269), provides the NRC's staff assessment of the challenges of incorporating diverse and flexible coping strategies and equipment (FLEX) into a PRA model in support of risk-informed decision-making in accordance with the guidance of RG 1.200, Revision 2 (ADAMS Accession No. ML090410014). Though implementation of FLEX procedures is cited in Exelon's December 13, 2018, letter as possible RMAs, the letter and other docketed information do not indicate if Byron and Braidwood have credited FLEX equipment or actions in the internal events, including internal flooding, or fire PRA models. As such, please address the following:

a. Discuss whether Exelon has credited FLEX equipment or FLEX mitigating actions into the Byron and Braidwood Station internal events, including the internal flooding, or fire PRA models. If not incorporated or the inclusion is not expected to impact the PRA results used in the RICT program, no additional response is requested.

b. If FLEX equipment or FLEX mitigating actions have been credited in the PRA, address the following, separately for the internal events, including internal flooding, and fire PRA:

    i. Summarize the supplemental equipment and compensatory actions, including FLEX strategies that have been quantitatively credited for each of the PRA models used to support this application. Include discussion of whether the credited FLEX equipment is portable or permanently installed equipment.

    ii. Discuss whether the credited equipment (regardless of whether it is portable or permanently-installed) are like other plant equipment (i.e. SSCs with sufficient plant-specific or generic industry data) and whether the credited operator actions are similar to other operator actions evaluated using approaches consistent with the endorsed ASME/ANS RA-Sa-2009 PRA standard.

    iii. If any credited FLEX equipment is dissimilar to other plant equipment credited in the PRA (i.e., SSCs with sufficient plant-specific or generic industry data), discuss the data and failure probabilities used to support the modeling and provide the rationale for using the chosen data. Discuss whether the uncertainties associated with the parameter values are in accordance with the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2.

    iv. If any operator actions related to FLEX equipment are evaluated using approaches that are not consistent with the endorsed ASME/ANS RA-Sa-2009 PRA Standard (e.g., using surrogates), discuss the methodology used to assess human error probabilities for these operator actions. The discussion should include:

1. A summary of how the impact of the plant-specific human error probabilities and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of the ASME/ANS RA-Sa-2009 PRA Standard were evaluated.

2. Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and if the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of the ASME/ANS RA-Sa-2009 PRA Standard.

3. If the procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

c. The ASME/ANS RA-Sa-2009 PRA standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 PRA Standard states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this Standard.

   Provide an evaluation of the model changes associated with incorporating mitigating strategies, which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.

d. Section 2.3.4 of NEI 06-09, Revision 0-A, states that PRA modeling uncertainties shall be considered in the application of the PRA base model results to the RICT program. The NRC SE for NEI 06-09 states that this consideration is consistent with Section 2.3.5 of RG 1.177 Revision 1. NEI 06-09, Revision 0-A further states that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties which could potentially impact the results of a RICT calculation. NRC staff notes that the impact of model uncertainty could vary based on the proposed RICTs. NEI 06-09 Revision 0-A also states that the insights from the sensitivity studies should be used to develop appropriate compensatory RMAs, including highlighting risk significant operator actions, confirming availability and operability of important standby equipment, and assessing the presence of severe or unusual environmental conditions. Uncertainty exists in modeling FLEX equipment and actions related to assumptions regarding the failure probabilities for FLEX equipment used in the model, the corresponding operator actions, and pre-initiator failure probabilities. Therefore, FLEX modeling assumptions can be key assumptions and sources of uncertainty for RICTs proposed in this application. In light of these observations:

   i. Describe the sensitivity studies that will be used to identify the RICTs proposed in this application for which FLEX equipment and/or operator actions are key assumptions and sources of uncertainty (e.g., use of generic industry data for non-safety related equipment). Explain and justify the approach used to perform the sensitivity studies.

ii. Describe how the results of the sensitivity studies which identify FLEX equipment and/or operator actions as key assumptions and sources of uncertainty will be used to identify RMAs prior to implementation of the RICT program, consistent with the guidance in Section 2.3.4 of NEI 06-09, Revision 0-A.

iii. Demonstrate the approaches described in items (i) and (ii) above using an example sensitivity study for the nominal configuration of a proposed RICT where the FLEX equipment and/or operator actions are identified as key assumptions and sources of uncertainty.

EGC RESPONSE to APLA RAI 04:

a. Special FLEX equipment and procedures are not currently credited in the internal events, internal flooding or fire PRA models. They are under consideration for addition to the ongoing periodic update model. When implemented, their incorporation will be performed in accordance with the ASME/ANS PRA Standard and NEI 06-09 Revision 0-A as appropriate.

b. Not applicable based on the response to item a above.

c. Not applicable based on the response to item a above.

d. Not applicable based on the response to item a above.

APLA RAI 05 - Real-Time Risk Model (RTR)

Regulatory Position 2.3.3 of RG 1.174, Revision 3, states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change. The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements.

Section 4.2 of NEI 06-09, Revision 0-A, describes attributes of the configuration risk management tool (CRM). A few of these attributes are listed below:

- Initiating events accurately model external conditions and effects of out-of-service equipment.
- Model translation from the PRA to a separate CRM tool is appropriate; CRM fault trees are traceable to the PRA. Appropriate benchmarking of the CRM tool against the PRA model shall be performed to demonstrate consistency.
- Each CRM application tool is verified to adequately reflect the as-built, as-operated plant, including risk contributors which vary by time of year or time in fuel cycle or otherwise demonstrated to be conservative or bounding.
- Application specific risk important uncertainties contained in the CRM model (that are identified via PRA model to CRM took benchmarking) are identified and evaluated prior to use of the CRM tool for RMTS applications.

- CRM application tools and software are accepted and maintained by and appropriate quality program.

- The CRM tool shall be maintained and updated in accordance with approved station procedures to ensure it accurately reflects the as-built, as-operated plant.

Exelon's letter dated December 13, 2018, Enclosure 8, describes the attributes of the RTR Model, or Braidwood and Byron's CRM tool, for use in RICT calculations. Exelon's letter explains that the internal flooding model is integrated into the internal events PRA model, but the fire PRA model is maintained as a separate model. Exelon's letter also describes several changes made to the internal events and fire PRA models to support calculation of configuration-specific risk and identifies approaches for ensuring the fidelity of the RTR to the PRAs including RTR maintenance, documentation of changes, and testing. With regards to development and application of the RTR model, provide the following:

a. Explain how any changes in success criteria based on seasonal variations are accounted for in the RTR Model for use in RICT calculations.

b. Confirm that out-of-service equipment will be properly reflected in the RTR Model initiating event models as well as in the system response models.

c. Describe the process that will be used to maintain the accuracy of any pre-solved cut sets with changes in plant configuration.

d. Describe the benchmarking activities performed to confirm consistency of the RTR model to base PRA model results, including periodicity of RTR updates compared to the base PRA model updates.

EGC RESPONSE to APLA RAI 05 Part a:

The outside air temperature is not modeled explicitly for BYR and BWD in the PRA. The room cooling calculations use conservative outside air temperatures. For example, the EDG calculation justifies the DGs can perform their function up to an outside air temperature of 95°F. In the event of extreme outside air temperatures, Operations will monitor local air temperature indications with more frequent rounds to maintain availability in accordance with site procedures. If the room temperature exceeds the operability limit, the equipment would be taken OOS in the RTR tool.

Operations verifies that the schedule used for the RICT calculation appropriately reflects these conditions in accordance with EGC Operations and Work Control procedures. This is consistent with the current (a)(4) process.

EGC RESPONSE to APLA RAI 05 Part b:

BYR and BWD PRA Models utilize system initiator event fault trees so equipment unavailabilities are captured explicitly in these system initiator fault trees.

EGC RESPONSE to APLA RAI 05 Part c:

Full PRA model quantifications will be used for each configuration. Pre-solved cutsets are only used for configurations that are identical to previously experienced configurations. For configurations without available configuration-specific cutsets the model will be quantified to produce cutsets for the specific RICT configuration. In accordance with EGC Risk Management procedures, when there are any changes in the underlying PRA model of record, the PRA Results database in the RTR tool will be updated to repopulate the configuration specific cutsets.

EGC RESPONSE to APLA RAI 05 Part d:

Every PRA model of record (MOR) Update results in an update to the RTR model in accordance with the EGC Risk Management FPIE update procedure and a pending update to the Fire PRA Update procedure. The RTR model documentation includes changes made to the MOR model files to work with the RTR model software (e.g., quantification settings) along with verification that results are consistent between the RTR and PRA zero maintenance results. The checks completed for this verification process are detailed in the Risk Management procedures. This is consistent with the current EGC 10 CFR 50.65 (a)(4) process. In addition, the RTR update for the MOR includes quantifying the RTR model for representative maintenance configurations and examining the results for appropriateness.

APLA RAI 06 - Identification of Compensatory Measures and RMAs

The NRC in its SE for NEI 06-09, Revision 0-A, states that the licensee should provide information that will describe the process to identify and provide compensatory measures and RMAs during extended CTs. Exelon, in its December 13, 2018, letter, Enclosure 12, identifies three kinds of RMAs (i.e., actions to provide increased risk awareness and control, reduction of the duration of maintenance activities, and reduction of the magnitude of risk increase). Enclosure 12 also provides examples of RMAs for an unavailable diesel generator, battery charger, RHR pump, and for loss of off-site power. Enclosure 12 does not describe what criteria or insights (e.g., important fire areas, important operator actions) are used to determine what RMAs to apply in specific instances. Therefore:

a. Describe the criteria and insights (e.g., important fire areas, important operator actions) that are used to determine the compensatory measures and RMAs to apply in specific instances.

b. Explain how RMAs are identified for emergent conditions in which the extent of condition evaluation for inoperable SSCs is not complete prior to exceeding the Completion Time to account for the increased possibility of a common cause failure (CCF). Include explanation of if and how these RMAs are different from other RMAs.

EGC RESPONSE to APLA RAI 06 Part a:

Risk Management Actions (RMAs) are compensatory measures to reduce risk. Determination of RMAs involves the use of both qualitative and quantitative considerations for the specific plant configuration and the practical means available to manage risk. The scope and number of RMAs developed and implemented are reached in a graded manner.

EGC Risk Management procedures contain guidance for development of RMAs in support of the RICT program. Development of RMAs considers those developed for other processes, such as the RMAs developed under the 10 CFR 50.65(a)(4) program and the protected equipment program. Operations has the final call on RMAs per the Operations RICT procedure. Additionally, Common Cause RMAs are developed to address the potential impact of common cause failures.

RMAs are identified based on the configuration-specific risk. There are three categories of RICT RMAs:

1) Actions to increase risk awareness and control, such as briefing of crews on risk important operator actions and procedures.

2) Actions to reduce the duration of maintenance activities, such as performing activities around the clock.

3) Actions to minimize the magnitude of the risk increase, such as protecting risk important equipment or minimizing fire risk in risk important rooms.

General RMAs are developed for input into the site-specific RICT system guidelines. These guidelines are developed using a graded approach. Consideration is given for system functionality. These RMAs include:

- Consideration of rescheduling maintenance to reduce risk
- Discussion of RICT in pre-job briefs
- Consideration of proactive return-to-service of other equipment
- Efficient execution of maintenance.

In addition to the RMAs developed qualitatively for the system guidelines, RMAs are developed based on the Real-Time Risk tool to identify configuration-specific RMA candidates to manage the risk associated with internal events, internal flooding, and fire events. These actions include:

- Identification of important equipment or trains for protection
- Identification of important Operator Actions for briefings
- Identification of key fire initiators and fire zones for RMAs in accordance with the site Fire RMA process
- Identification of dominant initiating events and actions to minimize potential for initiators
- Consideration of insights from PRA model cutsets, through comparison of importances.

Common cause RMAs are also developed to ensure availability of redundant SSCs, to ensure availability of diverse or alternate systems, to reduce the likelihood of initiating events that require operation of the out-of-service components, and to prepare plant personnel to respond

to additional failures. Common cause RMAs are developed by considering the impact of loss of function for the affected SSCs.

Examples of common cause RMAs include:

- Performance of non-intrusive inspections on alternate trains
- Confidence runs performed for standby SSCs
- Increased monitoring for running components
- Expansion of monitoring for running components
- Deferring maintenance and testing activities that could generate an initiating event which would require operation of potentially affected SSCs
- Readiness of operators and maintenance to respond to additional failures
- Shift briefs or standing orders which focus on initiating event response or loss of potentially affected SSCs.

EGC RESPONSE to APLA RAI 06 Part b:

Common cause RMAs are additional RMAs focused on ensuring availability of redundant components, ensuring availability of diverse or alternate systems, reducing the likelihood of initiating events to that require operation of the out-of-service components, and readiness of plant personnel to respond to additional failures.

In accordance with EGC procedures, for emergent conditions where the extent of condition is not completed prior to entering into the Risk Management Action Times or the extent of condition cannot rule out the potential for common cause failure, common cause RMAs are expected to be implemented to mitigate common cause failure potential and impact. These can include the pre-identified RMAs included in the system guidelines as discussed in the response to APLA RAI-06, item a, as well as additional common cause RMAs for the specific configuration. Appropriate RMAs, including both regular and common cause considerations, are developed for the specific configuration using the considerations specified in the response to APLA RAI-06, item a.

APLA RAI 07 - Evaluation of Common Cause Failures for Planned Maintenance

NEI 06-09, Revision 0-A, states that no common cause failure (CCF) adjustment is required for planned maintenance. The NRC SE for NEI 06-09, Revision 0, is based on conformance with RG 1.177, Revision 1. Specifically, the NRC SE, Section 2.2, states that, "specific methods and guidelines acceptable to the NRC staff are [...] outlined in RG 1.177 for assessing risk-informed TS changes." The NRC SE, Section 3.2, further states that compliance with the guidance of RG 1.174, Revision 1, and RG 1.177, Revision 1, "is achieved by evaluation using a comprehensive risk analysis, which assesses the configuration-specific risk by including contributions from human errors and common cause failures."

The guidance in RG 1.177, Revision 1, Section 2.3.3.1, states that, "CCF modeling of components is not only dependent on the number of remaining in-service components but is also dependent on the reason components were removed from service (i.e., whether for preventative or corrective maintenance)." In relation to CCF for preventive maintenance, the guidance in RG 1.177, Appendix A, Section A-1.3.1.1, states:

> If the component is down because it is being brought down for maintenance, the CCF contributions involving the component should be modified to remove the component and to only include failures of the remaining components (also see Regulatory Position 2.3.1 of Regulatory Guide 1.177).

According to RG 1.177, Revision 1, if a component from a CCF group of three or more components is declared inoperable, the CCF of the remaining components should be modified to reflect the reduced number of available components in order to properly model the as-operated plant.

a. Explain how CCFs are included in the PRA model (e.g., with all combinations in the logic models as different basic events or with identification of multiple basic events in the cut sets);

b. Explain how the quantification and/or models will be changed when, for example, one train of a 3×100 percent train system is removed for preventative maintenance and describe how the treatment of CCF meets the guidance in RG 1.177, Revision 1, or meets the intent of this guidance when quantifying a RICT.

EGC RESPONSE to APLA RAI 07 Part a:

In the BYR and BWD FPIE PRA models, CCF event probabilities are modeled and quantified using the "alpha factor" method described in NUREG/CR-5485 (Reference RAI-07a), with alpha factors from, or based on, updated CCF parameters on the NRC web site (Reference RAI-07b). Common cause basic events are explicitly modeled in the one-top fault tree, with each specific combination of events modeled in conjunction with the independent failure basic event. For example, consider a group consisting of components A, B, C, and D where failure of all four components were necessary to fail the PRA mitigation function. Failure of component A is modeled as an OR gate of the independent failure event for A, along with seven additional common cause basic events for combinations of AB, AC, AD, ABC, ABD, ACD, and ABCD.

In general, the base FPIE PRA model does not broadly adjust the modeled common cause basic events based on whether any components are out-of-service. However, because of the explicit modeling of the different common cause basic event combinations, the model implicitly captures the necessary logic. Following the ABCD example above, if component D were out of service, then the model would allow the combination of common cause event ABC with the basic event for D out-of-service to fail the mitigation function. The non-minimal combinations such as ABCD with the D out-of-service event may still appear in the cutsets if they are not subsumed but would represent only a slight conservatism.

While the modeling approach described above represents the general approach to common cause modeling, modeling of components in the Reactor Protection System and Engineered Safety Features Actuation System do include different modeling of maintenance configurations due to the unique impacts of testing configurations and potentially higher number of

components. For those systems, common cause events during test and maintenance conditions are developed as shown in the following examples.

Sensors/Process Logic/Bistables - CCF required for the function to be failed (e.g., if 2/4 required to process a trip, then 3/4 or more CCF will fail the function). All like CCF groups are typically combined into one basic event, then multiplied by the number of combinations (e.g., for 3/4 CCF, there are 4 possible combinations). When a channel is in maintenance, a different CCF basic event is used, which requires one less failure (and the group is one smaller too) if the channel in test is tripped, but the same number of failures if the channel is bypassed. Using the 3/4 CCF discussed above, if one channel is in test and tripped, then the group is now three, and all three remaining channels must fail for the function to fail (since the tested channel is in trip). Therefore, while in test (tripped), the CCF that would fail the function would be 3/3. If the channel in test is bypassed, then it cannot provide its signal, so the CCF term is now two or more of the remaining three channels.

Input Relays - Where the sensors, processors and bistables are common to both trains, the input relays are train specific. Therefore, for each train, there are common-cause groups similar to those developed for the sensor channels. However, common cause could occur across the trains (e.g., a CCF group of 6/8, which is 3/4 in train A and 3/4 in train B, would fail the function for both trains). Similar to the above, when a channel is in test/trip (not when a channel is in test/bypass), the number of components needed to fail the function changes.

EGC RESPONSE to APLA RAI 07 Part b:

The common cause events are not adjusted during quantification for planned maintenance.

Adjustments to the CCF grouping or CCF probabilities are not necessary when a component is taken out-of-service for preventative maintenance (PM). The component is not out-of-service for reasons subject to a potential common cause failure, and so the in-service components are not subject to increases in common cause probabilities.

The net failure probability for the in-service components includes the CCF contribution of the out-of-service component. This CCF contribution from the out-of-service component is conservatively retained two ways:

- The independent failure event used in the model includes both the independent and dependent failure probabilities.
- The CCF event probabilities that include the out-of-service component are retained.

As described in RG 1.177, Section A-1.3.2.2, the CCF term should be treated differently when a component is taken down for preventive maintenance than as described for failure of a component. For PMs, the common cause factor is changed so that the model represents the unavailability of the remaining component. In the example provided in the RG for a 2-train system, the CCF event can be set to zero for PMs. This is done so that the model represents the unavailability of the remaining component, and not the common cause multiplier. The EGC approach is conservative in that for a 2-train system, the CCF event is retained for the component removed from service. Likewise, for systems with three or more trains, the CCF events that are related to the out-of-service component are retained.

This is the same as the Calvert Cliff's approach described response to RAI 21 in reference APLA RAI 07c.

References:

RAI-07a: NUREG/CR-5485, Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment, Idaho National Engineering and Environmental Laboratory, June 1998

RAI-07b: https://nrcoe.inl.gov/resultsdb/ParamEstSpar/

RAI-07c: Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information License Amendment Request to Revise Technical Specifications to Adopt Risk Informed Completion Times TSTF-505, Revision 1, 'Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b'," dated June 21, 2018 (ADAMS Accession No. ML18172A145)

APLA RAI 08 - Evaluation of Common Cause Failure for Emergent Conditions

In its letter dated December 13, 2018, Exelon provides TS Administrative Section constraint d that states:

> For emergent conditions, if the extent of condition evaluation for inoperable structures, systems, or components (SSCs) is not complete prior to exceeding the ACTION allowed outage time, the RICT shall account for the increased possibility of common cause failure (CCF) by either:
>
> 1. Numerically accounting for the increased possibility of CCF in the RICT calculation; or
>
> 2. Risk Management Actions (RMAs) not already credited in the RICT calculation shall be implemented that support redundant or diverse SSCs that perform the function(s) of the inoperable SSCs, and, if practicable, reduce the frequency of initiating events that challenge the function(s) performed by the inoperable SSCs.

Regarding option 1 of constraint d, provide the following:

a. Describe and justify how the numerical adjustment for increased possibility of CCF will be performed, or

b. Confirm that numerically accounting for the increased possibility of CCF in the RICT calculation will be performed in accordance with RG 1.177, Revision 1.

EGC RESPONSE to APLA RAI 08 Part a:

Numerical adjustment of CCF events will not typically be performed for a RICT calculation. The proceduralized process is to complete an extent of condition assessment that precludes the possibility of CCF. If CCF cannot be ruled out, then risk management actions will be put in place to mitigate the potential for common cause. See the response to APLA RAI-06 for the process of identifying common cause RMAs.

EGC RESPONSE to APLA RAI 08 Part b:

As noted in the response to Part a above, CCF probabilities will normally not be adjusted for emergent failures. If a numeric adjustment is performed, the RICT calculation will be adjusted to numerically account for the increased possibility of CCF in accordance with RG 1.177, Revision 1, as specified in Section A-1.3.2.1 of Appendix A of the RG. Specifically, when a component fails, the CCF probability for the remaining redundant components will be increased to represent the conditional failure probability due to CCF of these components in order to account for the possibility the first failure was caused by a common cause mechanism.

APLA RAI 09 - PRA Modeling of Instrumentation and Controls

The proposed TS limiting conditions for operations (LCOs) in Exelon's December 13, 2018, letter include those related to instrumentation and controls (I&C). These include TS for the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS):

TS 3.3.1 RTS Instrumentation;
TS 3.3.2 ESFAS Instrumentation,
AND,
TS 3.3.5, Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation.

PRA technical acceptability attributes are provided in Section 2.3.4 of NEI 06-09, Revision 0-A, and in RG 1.200, Revision 2. The licensee has previously received approval for changes to its I&C completion times, bypass test times, and surveillance intervals consistent with guidance in Technical Specifications Task Force (TSTF) traveler TSTF-411 and TSTF-418. However, the licensee does not address whether the I&C are modeled in sufficient detail to support implementation of TSTF-505, Revision 2 (ADAMS Accession No. ML18183A493). The following additional information is requested:

a. Explain how instrumentation is modelled in the PRA. This should include, but not be limited to, the scope of the I&C equipment (e.g., channels, relays, logic) and associated TS functions for which a RICT would be applied, and PRA modeling of the I&C and functions including how these are modeled in sufficient detail and based on plant-specific data, etc.

b. Section 2.3.4 of NEI 06-09, Revision 0-A, states that PRA modeling uncertainties be considered in application of the PRA base model results to the RICT program. The NRC in its SE for NEI 06-09, Revision 0, states that this consideration is consistent with Section 2.3.5 of RG 1.177, Revision 1. NEI 06-09, Revision 0-A, further states that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties which could potentially impact the results of an RICT calculation and that sensitivity studies should be used to develop appropriate compensatory RMAs.

   Regarding digital I&C, the NRC staff notes the lack of consensus industry guidance for modeling these systems for plant PRAs to be used in risk-informed applications. In addition, known modeling challenges exist due to the lack of industry data for digital I&C components and the complexities associated with modeling software failures including common cause software failures. Given these needs and challenges, if the modeling of digital I&C systems are included in the RTR model, then address the following:

i. Provide the results of a sensitivity study on the SSCs in the RICT program demonstrating that the uncertainty associated with modeling the digital I&C system has inconsequential impact on the RICT calculations.

ii. Alternatively, identify which LCOs are determined to be impacted by the digital I&C system modeling for which RMAs will applied during a RICT. Explain and justify the criteria used to determine what level of impact to the RICT calculation required additional RMAs.

EGC RESPONSE to APLA RAI 09 Part a:

Much of the RPS and ESFAS instrumentation is modeled in detail in the PRA, including the necessary components from the sensors to the master and slave relays. Instrumentation is explicitly modeled, and the component failure rates are appropriate for the equipment. That is, component failure rates are based on accepted industry data, and updated with plant-specific data where required by the Standard (i.e., for risk-significant components).

TABLE APLA-RAI-09-1
REACTOR TRIP SIGNALS MODELED

REACTOR TRIP VARIABLE (ONLY MODELED SIGNALS ARE SHOWN)	SENSORS	PROCESS LOGIC MODULE	OUTPUT BISTABLE	SSPS INPUT RELAYS	S.S. LOGIC CARDS UNIVERSAL CARDS
	COMMON TO BOTH TRAINS (I.E., EACH CHANNEL INSTRUMENT LOOP IS COMMON TO BOTH TRAINS AND OUTPUTS TO BOTH TRAIN A AND B)			SEPARATE FOR EACH TRAIN (I.E., THERE ARE TWO ITEMS FOR EACH IDENTIFIED RELAYS OR LOGIC CARDS)	
Power Range High Flux (EF-21)	N41AB UIC[2] N42AB UIC N43AB UIC N44AB UIC	Summing Amplifier	NC41R NC42R NC43R NC44R	K147 K221 K302 K402	A404 (2/4)
Over Temperature ΔT (EF-28)	3 Hot Leg RTDs, 1 Cold Leg RTD and one pressure sensor	(Note 1)	TB411C TB421C TB431C TB441C	K112 K236 K323 K421	A313 (2/4)
Over Power ΔT (EF-28)	3 Hot Leg RTDs, 1 Cold Leg RTD and one pressure sensor	(Note 1)	TB411G TB421G TB431G TB441G	K111 K235 K322 K422	A312 (2/4)
Pressurizer Low Pressure (EF-23)	PT455 PT456 PT457 PT458	PY455B PY456B PY457B PY458B	PB455C PB456C PB457C PB458C	K129 K211 K343 K416	A417 (2/4)

[2] 2 sensors per channel - upper and lower; combined into one event.

TABLE APLA-RAI-09-1
REACTOR TRIP SIGNALS MODELED

REACTOR TRIP VARIABLE (ONLY MODELED SIGNALS ARE SHOWN)	SENSORS	PROCESS LOGIC MODULE	OUTPUT BISTABLE	SSPS INPUT RELAYS	S.S. LOGIC CARDS UNIVERSAL CARDS
	COMMON TO BOTH TRAINS (I.E., EACH CHANNEL INSTRUMENT LOOP IS COMMON TO BOTH TRAINS AND OUTPUTS TO BOTH TRAIN A AND B)			SEPARATE FOR EACH TRAIN (I.E., THERE ARE TWO ITEMS FOR EACH IDENTIFIED RELAYS OR LOGIC CARDS)	
Loss of Flow, Single RC Loop (EF-25)	FT414 FT415 FT416		FB414A FB415A FB416A	K125 K213 K312	A203 (2/3)->
	FT424 FT425 FT426		FB424A FB425A FB426A	K126 K214 K313	A206 (2/3)->
	FT434 FT435 FT436	N/A	FB434A FB435A FB436A	K127 K227 K314	A208 (2/3)->
	FT444 FT445 FT446		FB444A FB445A FB446A	K128 K228 K321	A218 (2/3)-> A308 (1/4)
Steam Generator Lo-Lo Water Level (EF-34)	LT556 LT519 LT518 LT517		LB556C LB519B LB518B LB517B	K150 K230 K331 K407	A203(2/4) ->
	LT529 LT557 LT528 LT527		LB529B LB557C LB528B LB527B	K114 K250 K332 K408	A206(2/4)->
	LT539 LT558 LT538 LT537	N/A	LB539B LB558CLB538B LB537B	K113 K255 K333 K409	A208(2/4)->
	LT559 LT549 LT548 LT547		LB559C LB549B LB548B LB547B	K121 K231 K334 K410	A218 (2/4)-> A316 (1/4)

TABLE APLA-RAI-09-1
REACTOR TRIP SIGNALS MODELED

REACTOR TRIP VARIABLE (ONLY MODELED SIGNALS ARE SHOWN)	SENSORS	PROCESS LOGIC MODULE	OUTPUT BISTABLE	SSPS INPUT RELAYS	S.S. LOGIC CARDS UNIVERSAL CARDS
	COMMON TO BOTH TRAINS (I.E., EACH CHANNEL INSTRUMENT LOOP IS COMMON TO BOTH TRAINS AND OUTPUTS TO BOTH TRAIN A AND B)			SEPARATE FOR EACH TRAIN (I.E., THERE ARE TWO ITEMS FOR EACH IDENTIFIED RELAYS OR LOGIC CARDS)	
Turbine Trip (Note 3) (EF-27)	1PS-EH024 1PS-EH026 1PS-EH028 1ZS-MS175 1ZS-MS176 1ZS-MS177 1ZS-MS178			K109 K241 K339 K108 K242 K338 K423	A403(2/3)-> and A404(2/2)-> A404(2/2)-> A403(2/2)
RCP Breaker Position (Note 4) (EF-26)	RCP 1A RCP 1B RCP 1C RCP 1D	N/A	N/A	K110 K238 K341 K414	A307(1/2)-> A215(1/2)-> A307(2/4)
Safety Injection Input from ESF	(See next table)				
Manual	(Note 5)				

TABLE APLA-RAI-09-1
REACTOR TRIP SIGNALS MODELED

REACTOR TRIP VARIABLE (ONLY MODELED SIGNALS ARE SHOWN)	SENSORS	PROCESS LOGIC MODULE	OUTPUT BISTABLE	SSPS INPUT RELAYS	S.S. LOGIC CARDS UNIVERSAL CARDS
	COMMON TO BOTH TRAINS (I.E., EACH CHANNEL INSTRUMENT LOOP IS COMMON TO BOTH TRAINS AND OUTPUTS TO BOTH TRAIN A AND B)			SEPARATE FOR EACH TRAIN (I.E., THERE ARE TWO ITEMS FOR EACH IDENTIFIED RELAYS OR LOGIC CARDS)	
For Note 1: Inputs to Over Temperature ΔT and Over Power ΔT					
RTD Location	RTD Amplifiers	Summing Amplifier	Summing Amplifier & Lead/Lag cards	OTDT Bistable	OPDT Bistable
Hot Leg 1 Hot Leg 2 Hot Leg 3 Cold Leg	TY411A1 TY411A2 TY411A3 TY411B	TY411A	TY411R L TY411K L TY411C	TB411C	TB411G
Hot Leg 1 Hot Leg 2 Hot Leg 3 Cold Leg	TY421A1 TY421A2 TY421A3 TY421B	TY421A	TY421R L TY421K L TY421C	TB421C	TB421G
Hot Leg 1 Hot Leg 2 Hot Leg 3 Cold Leg	TY431A1 TY431A2 TY431A3 TY431B	TY431A	TY431R L TY431K L TY431C	TB431C	TB431G
Hot Leg 1 Hot Leg 2 Hot Leg 3 Cold Leg	TY441A1 TY441A2 TY441A3 TY441B	TY441A	TY441R L TY441K L TY441C	TB441C	TB441G

Notes:

1. Refer to Table of Inputs to Over Temperature ΔT and Over Power ΔT, above. TY411A and TY411B input to TY411R that sends signal to TY411K then TY411C before the signal output to the bistables, TB411G (OPDT) and TB411C (OTDT). Also, TY411A gets its input from TY411A1, TY411A2 and TY411A3 before the signal is summed. Some of these components are either combined together into a single basic event, or the basic event represents more components than is listed in the table below.

2. Note Deleted.

3. For turbine trip, must have EITHER 2/3 EHC low pressure signals OR 4/4 (actually 2/2 and 2/2) valve limit switch signals successful.

4. For RCP trip, must have indication of at least 2 RCP breakers open via the breaker 52/a contacts.

5. For Manual reactor trip, one of two switches (RT1 or RT2) must function.

TABLE APLA-RAI-09-2
SAFETY INJECTION SIGNALS MODELED

SAFETY INJECTION VARIABLE (ONLY MODELED SIGNALS ARE SHOWN)	SENSORS	OUTPUT BISTABLE	SSPS INPUT RELAYS	S.S. LOGIC CARDS UNIVERSAL CARDS	S.S. LOGIC CARDS SAFEGUARDS OUTPUT CARDS
	COMMON TO BOTH TRAINS (I.E., EACH CHANNEL INSTRUMENT LOOP IS COMMON TO BOTH TRAINS AND OUTPUTS TO BOTH TRAIN A AND B)			SEPARATE FOR EACH TRAIN (I.E., THERE ARE TWO ITEMS FOR EACH IDENTIFIED RELAYS OR LOGIC CARDS)	
Pressurizer Low Pressure (EF-33)	PT455 PT456 PT457 PT458	PB455D PB456D PB457D PB458D	K131 K201 K344 K444	A315 (2/4) - A416(1/1)	A517 - A516
Low Steamline Pressure (EF-32)	PT516 PT515 PT514	PB516A PB515B PB514B	K417 K247 K133	A213(2/3) - A308(1/1)	
	PT526 PT525 PT524	PB526A PB525B PB524B	K317 K203 K134	A213(2/3) - A308(1/1)	A516
	PT534 PT536 PT535	PB534B PB536A PB535B	K119 K318 K248	A216(2/3) - A311(1/1)	
	PT544 PT546 PT545	PB544B PB546A PB545B	K118 K418 K204	A216 (2/3) - A311(1/1)	A517
High-1 Containment Pressure (EF-35)	PT936 PT935 PT934	PB936B PB935B PB934B	K217 K330 K430	A210(2/3)	A517
Manual	(Note 6)				

Note:
6. For Manual SI, one of two switches per train (SIA1 or SIA2, SIB1 or SIB2) must function.

Additional detailed information is documented in the FPIE PRA RPS & ESFAS System Notebook.

Some of the specific TS LCOs are represented by this detailed modeling. For those, the specific components would be modeled in the RTR model and the RICT calculated. For those LCOs that are not explicitly modeled, a surrogate event at a higher level in the fault tree logic is used to calculate a RICT which would be equal to or shorter than if the detailed modeling were explicitly included in the PRA model.

For example, TS 3.3.1.N, One Turbine Trip channel inoperable, is not explicitly modeled in the PRA. It is represented by surrogate PRA basic event 1RD05E-RTA--RBMM, REACTOR TRIP BREAKER RTA UNAVAILABLE DUE TO MAINTENANCE, which is a higher level failure in the fault tree model. In the example calculation for the LAR, this still results in reaching the 30-day backstop, so there is no impetus for developing more detailed modeling at this time.

EGC RESPONSE to APLA RAI 09 Part b:

No digital instrumentation or controls are being proposed for inclusion in the RICT program at this time.

APLA RAI 10 - Total Risk Estimates Against RG 1.174 Guidelines

RG 1.174 provides the risk acceptance guidance for total CDF (1E-04 per year) and LERF (1E-05 per year). In its letter dated December 13, 2018, Enclosure 4, Exelon shows that for Braidwood the total CDF for Unit 2 is 8.2E-05 per year and for Byron Unit 2 the CDF is 8.0E-05 per year). The NRC staff notes that the implementation items identified in Attachment 5 of Exelon's letter involve updates to the internal events and fire PRA models and the response to some of the RAIs may also involve updates to the internal events and fire PRA models. Accordingly, there appears to be a possibility that the PRA updates could increase the Byron and Braidwood CDF and LERF values above the RG 1.174 risk acceptance guidelines. Therefore, for the Byron and Braidwood Stations, either:

a. Demonstrate that after the internal events and fire PRA models are updated to execute implementation items in response to the RAIs that the total risk for each unit is recalculated from the updated models and confirmed to be in conformance with the RG 1.174 risk acceptance guidance (i.e., CDF < 1E-04 and LERF < 1E-05 per year).

b. Propose a mechanism ensuring that after the internal events and fire PRA models are updated to execute implementation items and in response to RAIs, the total risk for each unit is recalculated from the updated models and confirmed to be in conformance with risk acceptance guidance in RG 1.174 prior to the implementation of the RICT Program.

EGC RESPONSE to APLA RAI 10 Part a:

The BYR and BWD full power internal events PRA (FPIE) and fire PRA (FPRA) models that will be used for the RICT program incorporate changes necessary to support the use of the PRA model in implementation of the 50.69 program. The Fire PRA uses the same FPIE and internal flooding model, including resolution of all open F&Os as specified in the LAR implementation items. Since the same PRA model implementation items apply to the RICT program, the numerical results obtained with the models reflecting resolution of the 50.69 implementation items apply to both programs. The CDF and LERF results for the FPIE/internal flooding and FPRA models that address the implementation items are provided below, along with the total risk results for the modeled hazards.

TABLE APLA 10-1 BRAIDWOOD PRA-MODELED PLANT RISK RESULTS			
BRAIDWOOD	**INTERNAL EVENTS INCLUDING INTERNAL FLOOD**	**FIRE**	**PRA MODEL TOTAL (1)**
BW_U1_CDF	1.2E-05	6.2E-05	7.4E-05
BW_U1_LERF	7.2E-07	6.7E-06	7.4E-06
BW_U2_CDF	1.2E-05	6.4E-05	7.7E-05
BW_U2_LERF	7.2E-07	6.3E-06	7.1E-06

(1) Totals reflect rounding in the input values

TABLE APLA 10-2 BYRON PRA-MODELED PLANT RISK RESULTS			
BYRON	**INTERNAL EVENTS INCLUDING INTERNAL FLOOD**	**FIRE**	**PRA MODEL TOTAL (1)**
BY_U1_CDF	1.2E-05	6.5E-05	7.7E-05
BY_U1_LERF	6.3E-07	5.8E-06	6.4E-06
BY_U2_CDF	1.2E-05	6.6E-05	7.8E-05
BY_U2_LERF	6.3E-07	6.3E-06	6.9E-06

(1) Totals reflect rounding in the input values

As noted in the TSTF-505 LAR, a seismic risk model is not available for BWD or BYR. Estimates of seismic CDF for both stations were provided in Enclosure 4 of the LAR, using a method generally recognized as reasonable for obtaining such an estimate. The results reported in Enclosure 4 of the LAR are as follows:

BWD seismic CDF estimate: 4.2E-6/yr.
BYR seismic CDF estimate: 7.9E-6/yr.

Including these estimates in the totals shown above results in total CDF estimates for BWD and for BYR that are within the RG 1.174 guidance, i.e., CDF < 1E-4.

Estimates of seismic LERF for both stations were provided in Enclosure 4 of the LAR. As noted in NRC RAI APLB-01, the staff did not accept this approach as providing a sufficiently bounding estimate of seismic LERF for application in the RICT program. The response to RAI APLB-01 provides an alternate approach for obtaining a reasonably bounding seismic LERF penalty, and the associated seismic LERF penalty values are reported in the response to RAI APLB-01. However, the approach to estimating these values is judged to be inappropriate for obtaining a reasonably realistic representation of seismic LERF, as is needed for comparison to the RG 1.174 total LERF guidance. The conservatisms inherent in the approach used in the RAI APLB-01 response include assuming that the equipment required to respond to seismic events

for each plant would only have a seismic capacity equal to the IPEEE screening level of 0.3g, whereas a more realistic HCLPF for containment penetrations and containment isolation valves would be much higher and support lower conditional probabilities for containment isolation failure over the range of g-levels; and similarly that seismic capability of the containment for each plant is only credited to be the review level earthquake of 0.3g despite the IPEEE conclusions that there were no containment related seismic vulnerabilities.

Given the assessment in the respective plant IPEEEs that containment and containment isolation capabilities are robust, and that there are no containment or containment bypass seismic vulnerabilities, then in the absence of plant-specific information one approach to estimate seismic LERF is to use seismic capacity values for the containment from the NRC Risk Assessment of Operational Events (RASP) Handbook (Reference RAI 10-1). Tables 4-3 and 4-4 of Reference RAI 10-1 provide selected seismic capacities for various components and structures from two SPAR External Hazards models, and lists the following parameters for "Containment, Buildings:"

Table 4-3: Am = 1.1g, beta-r=0.3, beta-u=0.35
Table 4-4: Am = 1.1g, beta-r=0.2, beta-u=0.35.

Using these values with the same calculation approach described in the response to question APLB-01 (i.e., convolving the updated plant-specific seismic hazard for each plant with the IPEEE plant response HCLPF of 0.3g and with the RASP Handbook containment HCLPF) results in the following, for either case:

BWD seismic LERF estimate: 1.2E-6/yr.
BYR seismic LERF estimate: 2.5E-6/yr.

The EPRI Seismic PRA Implementation Guide (Reference RAI 10-2) also provides "representative" seismic capacity information for various components. Table H-1 of Reference 10-2 provides the following seismic capacity recommended values for "Containment, pre-stressed, shear:" Am = 2.0g, beta-r=0.3, beta-u=0.35. That table entry also notes "multiple other containment fragilities [from other sources] in the 3-8g range." Using these values with the same calculation approach described above results in the following:

BWD seismic LERF estimate: 2.8E-7/yr.
BYR seismic LERF estimate: 5.8E-7/yr.

Therefore, given the IPEEE conclusions regarding no seismic vulnerability for containment and containment isolation capability, and the best available information for the likely range of containment seismic capacity, it is reasonable to estimate seismic LERF for BWD as in the range of 3E-7 to 1E-6/yr., and seismic LERF for BYR as in the range of 6E-7 to 3E-6/yr.

Combining these seismic LERF estimates with the FPIE and Fire PRA LERF values shown in the tables above, the total LERF estimates for BWD and for BYR are within the RG 1.174 guidance of total LERF < 1E-5.

Note that, for the RICT program calculations, the seismic LERF penalty values described in the response to RAI APLB-01 will be used, not the estimates shown here.

Note also that the current Fire PRA risk results reflect current approved Fire PRA methods, including the updated ignition frequencies and non-suppression probabilities of NUREG-2169

(Reference RAI 10-3), and updated heat release rates of NUREG-2178 (Reference RAI 10-4). Further refinements of these methods are in development (e.g., NUREG-2178, Volume 2 (Reference RAI 10-5) and NUREG-2230 (Reference RAI 10-6)) which are expected to increase realism and reduce conservatism in Fire PRA analysis based on the currently approved methods. Future incorporation of these methods into the BWD and BYR Fire PRAs along with further Fire PRA refinement activities is expected to reduce the fire risk (CDF and LERF) and ensure that the total risk for BWD and BYR remains below the RG 1.174 guidance.

EGC RESPONSE to APLA RAI 10 Part b:

Prior to implementation of the RICT program, a proceduralized check of the overall PRA results against the RG 1.174 guidance will be performed. This check is also being incorporated into the EGC PRA Model update procedures.

References:

RAI 10-1: "Risk Assessment of Operational Events (RASP) Handbook, Vol 2 - External Events," Rev 1.02, Nov 2017

RAI 10-2: EPRI 3002000709, "Seismic PRA Implementation Guide," Electric Power Research Institute, Final Report, December 2013

RAI 10-3: "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009," NUREG-2169, EPRI 3002002936, US NRC and Electric Power Research Institute, January 2015

RAI 10-4: "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE) — Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume," NUREG-2178, Volume 1, EPRI 3002005578, US NRC and Electric Power Research Institute, April 2016

RAI 10-5: "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire, Volume 2: Fire Modeling Guidance for Electrical Cabinets, Electric Motors, Indoor Dry Transformers, and the Main Control Board," NUREG-2178, Volume 2, EPRI 3002016052, Draft for Comment, April 2019

RAI 10-6: "Methodology for Modeling Fire Growth and Suppression Response of Electrical Cabinet Fires in Nuclear Power Plants," NUREG-2230, EPRI 3002016051, Draft for Comment, April 2019

APLA RAI 11 - Potential Loss of Function Conditions

TSTF-505, Revision 2, July 2, 2018 (ADAMS Accession No. ML18183A493) does not allow for TS loss of function conditions (i.e., those conditions that represent a loss of a specified safety function or inoperability of all required trains of a system required to be OPERABLE) in the RICT program. Address the following:

a. LCO 3.7.9 Ultimate Heat Sink (UHS)

A number of aspects of LCO 3.7.9 (Ultimate Heat Sink) Condition E (One or more basin levels(s) <60%) for the Byron Station as presented in Exelon's letter dated December 13, 2018, Enclosure 1, Table E1-1 are not clear.

First, LCO 3.7.9, Condition E appears to represent a TS loss of function condition with both Ultimate Heat Sink (UHS) basins inoperable (both basins levels are < 60%). Also, if the configuration at the site is such that one basin is aligned to one reactor unit and the other basin is aligned to the other reactor unit, then it is possible that the inoperability of one basin might also represent a TS loss of function if the basin for the opposite unit cannot be realigned in time to respond to a particular event.

Second, the basis for the PRA success criteria for the basins is not clear. Exelon, in its letter dated December 13, 2018, Enclosure 1, Table E1-1, states that the PRA success criteria is "one basin with the ability to receive makeup from: a single [Essential Service Water] SX makeup pump OR single [Circulating Water] CW makeup pump OR both Deep Well pumps." It is not clear what water sources feed the SX and CW makeup paths to the basins to provide an equivalent or greater sink heat capacity as the heat sink capability provided by a basin at 60% level or how this equivalency was established.

In addition, the surrogate modeling that will be used in the PRA models to reflect the unavailability of the basins is not clear. Exelon, in its letter dated December 13, 2018, indicates that the basins are not modelled in the PRA, and so surrogate modelling will be used in RICT calculations to fail all makeup sources to the basins for Condition E. This approach appears to eliminate any credit for UHS for this configuration (i.e., LCO 3.7.9, Condition E).

Considering these observations, address the following for the Byron Station:

i. Justify that LCO 3.7.9, Condition E does not represent TS loss of function. Include discussion of the case in which both basins are TS inoperable and the case in which one basin is TS inoperable. Also, include discussion about how the basins are aligned to each reactor unit.

ii. If in response to item (i) above, it cannot be justified that LCO 3.7.9, Condition E does not represent TS loss of function, then remove this LCO condition from the RICT program and provide an updated TS markup.

iii. If in response to item (i) above, it can be justified that LCO 3.7.9, Condition E does not represent TS loss of function, then:

    1. Describe the water sources for the SX and CW makeup paths to the basins that are modelled in PRAs and provide the basis used to determine that those makeup sources provide equivalent or greater sink heat capacity as that provided by a basin at 60% level. Include explanation of what the CW system is and what function it provides.

2. Describe the surrogate modelling that will be used in the PRA to reflect the unavailability of the basins for LCO Condition E. Include clarification of whether the use of the surrogate will conservatively result in loss of UHS for this configuration.

b. LCO 3.4.11.B Pressurizer Power Operated Relief Valves (PORV)

Exelon, in its letter dated December 13, 2018, Enclosure 1, Table E1-1, indicates that the design success criteria for the PORV function is two of two operable PORVs. LCO 3.4.11 (PORVs) Condition B (One PORV inoperable and not capable of being manually cycled) appears to represent a TS loss of function condition for both the Byron and Braidwood Stations. One operating PORV is not enough to satisfy the design basis success criteria. Therefore, for the Byron and Braidwood Stations:

i. Justify that LCO 3.4.11, Condition B does not represent TS loss of function. Include discussion of whether there are other safety related SSCs that can provide the safety function provided by the PORVs.

ii. If in response to item i. above, it cannot be justified that LCO 3.4.11, Condition B does not represent TS loss of function, then remove this LCO condition from the RICT program and provide an updated TS markup.

c. LCO 3.7.9.B One required Service Water Cooling Tower (SXCT) fan inoperable

Exelon, in its letter dated December 13, 2018, Enclosure 1, Table E1-1 for the Byron Station, states that the design criteria for this condition is "6-8 out of 8 fans depending on [Service Water] SX pump discharge water temperature and if the SX trains on each unit are cross-tied" and "8/8 fans no crosstie." Accordingly, the success criteria varies depending on SX pump discharge water temperature and whether the SX trains have been cross-tied. Furthermore, it appears that one required SXCT fan inoperable can create a TS loss of function depending on the SX pump discharge water temperature or if the SX trains are not cross-tied. Therefore, for the Byron and Braidwood Stations:

i. Justify that LCO 3.7.9, Condition B does not represent TS loss of function.

ii. If in response to part (i) above, it cannot be justified that LCO 3.7.9, Condition B does not represent TS loss of function, then remove this LCO condition from the RICT program.

<u>EGC RESPONSE to APLA RAI 11 Part a:</u>

After further evaluation, EGC has determined to remove Condition E of LCO 3.7.9 from the scope of the RICT program. Attachment 2 provides revised TS markup pages for BYR LCO 3.7.9.

<u>EGC RESPONSE to APLA RAI 11 Part b:</u>

Table E1-1 incorrectly provided a design success criterion of TWO PORVs. Per the Technical Specification Bases 3.4.11, only ONE PORV is required to meet design basis requirements. Therefore, a single inoperable PORV does not represent a Loss of Function condition, and Table E1-1 will be revised to reflect the correct design basis requirement.

EGC RESPONSE to APLA RAI 11 Part c:

After further evaluation, EGC has determined to remove Condition B of LCO 3.7.9 from the scope of the RICT program. Attachment 2 provides revised TS markup pages for BYR LCO 3.7.9.

PRA External Hazards

APLB Question 01 – Bounding Seismic Risk Analysis

Section 2.3.1, Item 7, of NEI 06-09, Revision 0-A (ADAMS Accession No. ML12286A322), states that the "impact of other external events risk shall be addressed in the RMTS program," and explains that one method to do this is by "performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT." The NRC staff's safety evaluation for NEI 06-09 (ADAMS Accession No. ML071200238) states that "[w]here [probabilistic risk assessment] PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT."

Seismic PRAs are not available for either Braidwood (BWD) or Byron (BYR). Exelon, in its letter dated December 13, 2018, Enclosure 4, Section 3, states that a seismic core damage frequency (CDF) and large early release frequency (LERF) "penalty" was determined separately for Braidwood and Byron for this application using the recent seismic hazard curves developed in response to Recommendation 2.1 of the Near-Term Task Force (NTTF) (ADAMS Accession No. ML14091A005 and ML14091A010, respectively).

Details of the approach for determining the seismic "penalty" are provided in Section 3 of Enclosure 4 to Exelon's letter dated December 13, 2018. Exelon calculated the seismic LERF using the ratio between LERF and CDF, based on the internal events, including internal flooding. Exelon explained that the ratio was adjusted by removing the risk contribution of certain accident scenarios because they would not be expected to be induced by a seismic event. In Section 3 of Enclosure 4 to its letter dated December 13, 2019, Exelon stated that the chosen conditional large early release probability (CLERP) value for seismic events was "adequately conservative." As noted earlier the NEI 06-09, Revision 0-A as well as the corresponding NRC staff SE calls for a "bounding analysis." In addition, NRC staff has generically observed that LERF-to-CDF ratio for seismic events can be significantly higher than the LERF-to-CDF ratio for internal events due to the unique nature of seismically-induced failures. It is unclear that the selected CLERP of 15% can be considered as a bounding value for use in the RICT calculation.

a. Justify that the seismic LERF "penalties" provided in the submittal to support RICT calculations for the Byron and Braidwood are bounding. Include the rationale that deriving seismic LERF to CDF ratio using the internal events LERF to CDF ratio is bounding for seismically induced events, given that internal events random failures do not capture seismically-induced failures that may uniquely contribute to LERF.

b. If the approach to estimating seismic LERF cannot be justified as bounding for this application in response to part (a) above, then provide, with justification, the bounding seismic LERF "penalties" for use in calculating the proposed RICTs for Byron and Braidwood.

EGC RESPONSE to APLB Question 01 Part a:

Throughout NEI 06-09 Rev 0-A and the NRC SE for that document, reference is made to either a "bounding" or "conservative" analysis, or sometimes to a "reasonable bounding analysis", as being acceptable to account for risk for external hazards when a PRA model is not available. The references to estimation of a seismic LERF contribution for the RICT program as "bounding" should more accurately refer to this as a "conservative" analysis that uses an estimated averaged seismic conditional large early release probability (SCLERP) to determine a seismic LERF that is then conservatively used in RICT assessments. A truly "bounding" estimate for seismic LERF would require assuming SCLERP = 1.0, which is neither reasonable nor realistic.

In the absence of a seismic PRA for BYR and BWD, the approach initially used for the LAR to estimate SCLERP was to use the internal events and internal flooding PRAs to derive a CLERP value for each initiating event (i.e., large early release frequency for the initiating event divided by core damage frequency for the initiating event) in those PRAs for all initiating events other than direct containment bypass events. Although this approach attempted to reflect plant-specific considerations in the seismic LERF evaluation, it could not be shown to be sufficiently conservative in addressing the influence of seismic-induced failures, so the estimation of the average SCLERP may not be sufficiently conservative. The response to Part (b) provides an alternate assessment of a conservative CLERP value.

EGC RESPONSE to APLB Question 01 Part b:

In the absence of a seismic PRA or other detailed current seismic evaluation, the following approach will be used to estimate seismic LERF. A review of plant specific and generic information on LERF contributors identified for internal events and likely to be important for seismic events was performed. In the BWD and BYR internal events PRA, LERF is associated with core damage sequences resulting from Interfacing System Loss of Coolant Accidents (ISLOCAs), Steam Generator Tube Ruptures (SGTR), and failures of containment isolation. Information from recent PWR SPRAs indicates that steam generator tubes are judged not to be vulnerable to seismic events based on their ductile materials. ISLOCA is not anticipated to be a significant contributor to seismic LERF because ISLOCA scenarios typically involve failures of valve and check valve internals, and the valves in potential ISLOCA pathways generally have high seismic capacity, so these failure modes would not be contributors to seismic LERF.

Failure of containment isolation has been identified in SPRAs as a contributor to seismic LERF. While seismic failures of containment isolation have some degree of correlation with seismic CDF failures, containment isolation failures, typically associated with failures of valves, would be significantly uncorrelated with likely dominant seismic CDF failures. Regarding any potential unique seismic impacts, BWD and BYR are designed so that most SSCs important to containment isolation would fail in a desirable (isolated) state during an earthquake. Most containment isolation pathways at BWD and BYR are equipped with combinations of check valves, which are seismically rugged components, and air operated valves (AOV) that fail closed on loss of support and are also generally seismically rugged components. Certain containment isolation motor operated valves (MOV) require actuation signals and power for

closure, but most such valves would normally be already closed at the time of the earthquake. The BYR and BWD internal events PRA evaluated all containment isolation pathways and, after screening for potential to be LERF contributors, identified only two related pathways with potential seismic dependencies (i.e., failures of motive power or actuation signals). These are the containment recirculation sump suction lines, which rely on MOVs powered by safety-related 480V AC power. However, these valves would be closed at the time of the earthquake, so that seismically-induced failures are not of concern for containment isolation failure contribution to seismic LERF.

Given the above, the ability to isolate containment following a seismic event is not overly dependent on SSCs that might be vulnerable to seismic failure. The BYR and BWD IPEEEs (Reference 1, 2) confirmed, in Section 3.4, that the equipment required to respond to seismic events could be screened at the IPEEE screening level of 0.3g based on the walkdowns conducted. A more realistic HCLPF for containment penetrations and containment isolation valves would likely be much higher and support lower conditional probabilities for containment isolation failure over the range of g-levels.

The seismic capability of the containment for each plant was evaluated in the respective IPEEE studies. Those studies (References 1, 2) concluded, in Section 3.4.1, that there were no containment related seismic vulnerabilities, and describe the containments and associated interfaces (penetrations) as able to be screened at the review level earthquake of 0.3g established for the IPEEE review.

Therefore, an estimate of seismic LERF can be obtained by convolving the plant seismic core damage estimate developed for the LAR with an assumed containment integrity HCLPF to estimate seismic LERF. That is, the seismic LERF can be estimated by convolving the plant seismic hazard with the plant limiting HCLPF for core damage and the plant limiting HCLPF for containment integrity. Although it is believed that containment capability is significantly greater than the assessed value from the IPEEE, the IPEEE screening level can be used as a conservative (and likely bounding) value to estimate seismic LERF.

Applying this approach results in a seismic LERF estimate of 2.1E-6/yr for each BWD unit, and a seismic LERF estimate of 4.3E-6/yr for each BYR unit. These values will be used in the RICT program instead of the seismic LERF values stated in the LAR.

As noted in the response to Part a, the approach taken to estimate seismic LERF is conservative rather than truly bounding. Adequate conservatism for the RICT application derives from the proposed approach to apply the total estimated annual seismic LERF as a delta SLERF in each RICT calculation, regardless of the duration of the completion time. The total estimated annual seismic CDF and LERF are applied starting at time zero for each RICT calculation. Given that the maximum (backstop) completion time is 30 days, use of the annual seismic LERF (and seismic CDF) in the RICT calculations introduces a factor of at least 12 conservatism in the seismic contribution to each RICT calculation.

References:

1. "Byron Nuclear Power Station Units 1 and 2 Individual Plant Examination of External Events for Severe Accident Vulnerabilities Submittal Report," Commonwealth Edison Company, December 1996

2. "Braidwood Nuclear Power Station Units 1 and 2 Individual Plant Examination of External Events for Severe Accident Vulnerabilities Submittal Report," Commonwealth Edison Company, June 1997

APLB Question 02 – Extreme Winds Analysis

Section 2.3.1, Item 7, of NEI 06-09, Revision 0-A, states that the "impact of other external events risk shall be addressed in the RMTS program," and explains that one method to do this is by documenting prior to the RMTS program that external events that are not modeled in the PRA are not significant contributors to configuration risk. The NRC staff's SE for NEI 06-09 (ADAMS Accession No. ML071200238) states that, "[o]ther external events are also treated quantitatively, unless it is demonstrated that these risk sources are insignificant contributors to configuration-specific risk." Section 1.2.5 of Regulatory Guide (RG) 1.200, "*An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities*," Revision 2 (ADAMS Accession No. ML090410014), states that the contribution of many external events to CDF and LERF can be screened out "(1) if it meets the criteria in NRC's 1975 Standard Review Plan (SRP) or later revision; or (2) if it can be shown using a demonstrably conservative analysis that the mean value of the design-basis hazard used in the plant design is less than 10^{-5} per year and that the conditional core damage probability is less than 10^{-1}, given the occurrence of the design-basis-hazard event; or (3) if it can be shown using demonstrably conservative analysis that the CDF is less than 10^{-6} per year." The screening criteria listed in Section 1.2.5 of RG 1.200 are consistent with those in Section 6-2.3 of the 2009 American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard (RA-Sa-2009), "*Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications.*"

In Section 4 of Enclosure 4 to its letter dated December 13, 2018, Exelon addresses the risk from extreme winds. The licensee stated that the tornado generated missile damage is evaluated using a plant specific TORMIS analysis and concluded that the tornado missile hazard can be screened out from consideration. However, a recent study mentioned in its December 13, 2018, letter (Reference 24 of Enclosure 4) Exelon shows that, for Byron, certain combinations of unavailable components and/or trains could result in CDF and/or LERF exceeding the screening criteria of CDF=10^{-6} per year and LERF=10^{-7} per year. Therefore, conservative penalty factors are developed to account for tornado missile risk in the RICT calculations, as ΔCDF = 5×10^{-6} /yr and ΔLERF = 1×10^{-7} /yr based on that study. It is unclear to the NRC staff how the bounding values were determined for the tornado missile risk "penalty" at Byron.

a. Discuss the approach for determination of the tornado missile risk "penalty" for Byron and justify that the penalty is bounding for calculating the proposed RICTs.

b. Provide justification for the lack of a tornado missile risk "penalty" for calculating the proposed RICTs at Braidwood.

EGC RESPONSE to APLB Question 02 Part a:

The BYR TORMIS analysis (Reference 17 in Enclosure 4 to the TSTF-505 LAR) made use of Boolean logic to determine that the total damage frequency of unprotected SSCs at BYR is less than 1E-6/yr. Specifically, the damage frequency for the service water cooling towers (SXCT)

was calculated to determine the loss of cooling function as opposed to total arithmetic damage frequency. The assumptions in the TORMIS Boolean logic supported design basis considerations, but not necessarily the success criteria used in the PRA or the potential configurations that could be entered for a RICT. Therefore, the total damage frequency from TORMIS could not be directly correlated to CDF. However, the information was used to support a conservative risk analysis that (1) confirmed the total CDF is less than 1E-6/yr and (2) calculated penalty factors to apply to RICT calculations.

Using the data in the BYR TORMIS analysis, tornado-missile failure events for SXCT failures (individual and combinations) were used in a risk analysis to calculate CDF for scenarios involving tornado missile SXCT failures. The risk analysis was based on the BYR Full Power Internal Events (FPIE) PRA model, assuming a dual unit Loss of Offsite Power with no recovery or repair. A conservative estimate of the average maintenance CDF is less than 7E-7/yr, accounting for tornado missile failures of all combinations of SXCTs.

The CDF for various maintenance configurations was calculated, accounting for tornado missile failures of the SXCT. The configurations included the unavailability of individual and multiple SXCT cells and other risk significant SSCs. The highest CDF calculated was approximately 3E-6/yr, resulting in a ΔCDF of approximately 2.3E-6/yr (3E-6 – 7E-7 = 2.3E-6). This was conservatively rounded up to 5E-6/yr. The maximum ΔLERF for maintenance configurations was estimated to be approximately 4E-8/yr, which was rounded up to 1E-7/yr.

EGC RESPONSE to APLB Question 02 Part b:

As described in the TSTF-505 LAR, the BWD TORMIS analysis provides the basis for screening tornado missile risk at BWD. The BWD TORMIS analysis used the total arithmetic sum of the damage frequency for all unprotected SSCs. This is different from the BYR analysis, which made use of Boolean logic to determine the total damage frequency of unprotected SSCs (see response to part a). Since the total arithmetic sum of unprotected SSC damage frequencies is less than 1E-6/yr at BWD, total and configuration-specific CDF is less than 1E-6/yr. Therefore, no penalty factor is required for BWD RICT calculations.

ATTACHMENT 2
Proposed Technical Specifications Changes for Byron Station, Units 1 and 2

Byron Station, Units 1 and 2

Facility Operating License Nos. NPF-37 and NPF-66

Mark-up of Technical Specifications Pages

3.7.9-1
3.7.9-2

NO CHANGES TO BYRON LCO 3.7.9 WITH TSTF-505

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE and the required SX cooling tower (SXCT) fans shall be OPERABLE and operating as specified in Table 3.7.9-1 or Table 3.7.9-2.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more OPERABLE SXCT fan(s) not running in required high speed as required by Table 3.7.9-1 or Table 3.7.9-2.	A.1	Initiate actions to operate OPERABLE SXCT fan(s) in high speed.	Immediately
B. One required SXCT fan inoperable.	B.1	Verify OPERABLE SXCT fans are capable of being powered by an OPERABLE emergency power source.	1 hour
	AND		
	B.2	Restore required SXCT fan to OPERABLE status.	72 hours ~~RICT INSERT~~

(continued)

NO CHANGES TO BYRON LCO 3.7.9 WITH TSTF-505

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Outside air wet bulb temperature > 76°F. AND Any electrical division not capable of providing power to at least one OPERABLE SXCT fan.	C.1	Verify OPERABLE SXCT fans are capable of being powered by an OPERABLE emergency power source.	1 hour
	AND		
	C.2	Restore SXCT fan configuration such that each electrical division is capable of providing power to at least one OPERABLE SXCT fan.	72 hours
D. SX pump discharge water temperature > 96°F.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours
E. One or more basin level(s) < 60%.	E.1	Restore both basin levels to ≥ 60%.	6 hours

~~RICT INSERT~~

(continued)