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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: Supplemental 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)

4) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "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'," dated September 5, 2019 (ADAMS Accession No. ML19248C699)

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).

In EGC letter dated September 5, 2019 (Reference 4), a partial response to the requested information was provided. The Attachment to this letter provides the responses to APLA RAI 02 and APLA RAI 03.

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 23rd day of September 2019.

Respectfully,

Dwi Murray
Sr. Manager – Licensing
Exelon Generation Company, LLC

Attachment: Response to Request for Additional Information

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
Response to Request for Additional Information

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APLA RAI 02 - Key Assumptions and Sources of Uncertainty

Regulatory Position C of RG 1.174, Revision 3 (ADAMS Accession Number ML17317A256) states:

> "In risk-informed decisionmaking, licensing basis changes are expected to meet a set of key principles… In implementing these principles, the staff expects the following… Uncertainty receives appropriate consideration in the analyses and interpretation of findings… NUREG-1855 provides acceptable guidance for the treatment of uncertainties in risk-informed decision making"

NUREG-1855, Revision 1 (ADAMS Accession No. ML17062A466) provides guidance on screening sources of uncertainty and determining those that are key sources of uncertainty for the application. NUREG-1855, Revision 1 identifies Electric Power Research Institute (EPRI) Topical Report (TR) 1016737[1] and EPRI TR 1026511[2] as providing additional guidance for identifying and characterizing key sources of uncertainty.

Section 2.3.4 of Nuclear Energy Institute (NEI) 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines," Revision 0-A (ADAMS Accession No. ML12286A322), states that PRA modeling uncertainties be considered in application of the PRA base model results to the RICT program. The NRC Safety Evaluation (SE) for NEI 06-09, Revision 0, states that this consideration is consistent with Section 2.3.5 of RG 1.177, Revision 1 (ADAMS Accession No. ML100910008). 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. These sensitivity studies should be used to develop appropriate compensatory Risk Management Actions (RMAs) such as highlighting risk significant operator actions, confirming availability and operability of important standby equipment, and assessing the presence of severe or unusual environmental conditions.

In its December 13, 2018, letter Exelon states that the internal events PRA uncertainty analysis was performed based on guidance from NUREG-1885, Revision 1, (ADAMS Accession No. ML17062A466) and EPRI Topical Report (TR) 1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments" dated 2008. Exelon indicates that key plant-specific assumptions and modeling uncertainties from the internal events PRA documentation were considered, as well as generic sources of uncertainty from EPRI TR 1016737. However, in its letter dated December 13, 2018, as supplemented by letter dated June 19, 2019 [sic], Exelon does not discuss the sources of modeling uncertainty that were identified for the fire PRA or explain whether both plant-specific and generic modeling issues were considered. The NRC staff notes that generic modelling uncertainties for fire PRA are identified in EPRI TR 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty", dated 2012 and cited by NUREG-1855, Revision 1.

In addition, based on RG 1.174 and Section 6.4 of NUREG-1855, Revision 1, for a Capability Category II risk evaluation, the mean values of the risk metrics (total and incremental values) need to be compared against the risk acceptance guidelines. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on

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[1] Available from EPRI, 5637 Gulfstream Row, Columbia, MD 21044

the PRA input parameters and model uncertainties explicitly reflected in the PRA models. In general, the point estimate core damage frequency (CDF) and large early release frequency (LERF) obtained by quantification of the cut set probabilities using mean values for each basic event probability does not produce a true mean of the CDF/LERF. Under certain circumstances, a formal propagation of uncertainty may not be required if it can be demonstrated that the state of knowledge correlation (SOKC) is unimportant (i.e., the risk results are well below the acceptance guidelines). Enclosure 4 of Exelon's December 13, 2018, letter shows that for the Braidwood Station, Unit 2, the total CDF is 8.2E-05 per year and for Byron Station, Unit 2, the CDF is 8.0E-05 per year. Therefore, an increase in CDF and LERF due to SOKC could possibly impact the conclusions of this application.

Considering the observations above, address the following for the Byron and Braidwood Stations:

a. Describe the process used to identify and evaluate key assumptions and sources of model uncertainty for the fire PRA. Address the following in the response:

    i. Discuss how a comprehensive list of plant-specific and generic industry key assumptions and sources of uncertainty for the fire PRA were identified as a starting point for this evaluation.

    ii. Explain how the comprehensive list of key assumptions and sources of uncertainty sources were screened to a list of uncertainties that were specifically evaluated for their impact on the RICT application.

    iii. Explain what criteria or what additional analysis was used to evaluate the impact of the key assumptions and sources of uncertainty on the RICT application.

    iv. Describe how the evaluation process is consistent with guidance in NUREG-1855, Revision 1, or other NRC-accepted processes.

b. In accordance with the process described in NUREG-1855, Revision 1, describe any additional sources of fire PRA model uncertainty and related assumptions relevant to the application that were not provided in Exelon's letter dated December 13, 2018, Enclosure 9, and describe their impact on the application results.

c. For any additional sources of model uncertainty and related assumptions identified in item b, above:

    i. Provide qualitative or quantitative justification that these uncertainties and assumptions do not challenge the RG 1.174 risk acceptance guidelines.

    ii. Justify that these key uncertainties and assumptions have no impact on the RICT calculations or, if determined to have a significant impact, consistent with the guidance in NEI 06-09-A, discuss the RMAs for each key uncertainty and assumption that will be implemented to minimize their potential adverse impact.

d. Clarify whether the total CDF and LERFs values presented in Exelon's December 13, 2018, letter, Enclosure 5, are consistent with guidance in NUREG-1855, Revision 1 stipulating that the risk estimates should be based on mean values from parametric uncertainty analysis results that considers the SOKC for internal events and fire parameters. If risk values were not estimated in accordance with guidance in NUREG-1855, Revision 1 as described above, then confirm the RG 1.174 total risk acceptance guidelines are still met as a part of RAI 10, below.

EGC Response to APLA RAI 02, Part a.i:

The initial assessment for the fire PRA in the license amendment request (LAR) submitted December 13, 2018 (Reference 1) for Byron and Braidwood was based on the 16 tasks from NUREG/CR-6850 (EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," September 2005. Subsequent to that analysis, a detailed assessment of the generic industry potential sources of fire PRA model uncertainty from Appendix B of EPRI TR 1026511 was also performed. This did not identify any additional potential sources of uncertainty for the applications other than those summarized by the 16 tasks from NUREG/CR-6850 in Enclosure 9 of the LAR.

EGC Response to APLA RAI 02, Part a.ii:

The list of potentially important assumptions and related sources of model uncertainty as noted in the response to Part a.i above was screened based on a review of each topic and the manner in which the fire PRA incorporates the applicable guidance. The type of approach utilized in implementing each topic in the fire PRA (e.g., use of a consensus approach, or other applicable guidance), and the level of detail included in the fire PRA are means used to determine and screen potential impacts on the application. The unscreened assumptions and sources of model uncertainty represent those that may be key to the application.

EGC Response to APLA RAI 02, Part a.iii:

The criteria used to evaluate the key assumptions and sources of uncertainty include an assessment to determine whether each potential unscreened item noted in response to Part a.ii above would challenge the acceptance guidelines for RICT implementation. It is expected that reasonable changes to the assumptions associated with potential sources of uncertainty could lead to a shift in the calculated delta risk for the RICT application but would tend to not totally shift the dominant contributors for the configurations. In the case of RICT, appropriate compensatory measures to address the dominant contributors for the configuration will be in place prior to the risk management action time (RMAT) being exceeded and for the remaining duration of the RICT configuration to help to minimize the risk. RMAs will be developed as described in Enclosure 12 of the LAR, consistent with the guidance in NEI 06-09, Revision 0-A, using insights from the PRA models and other good practices (e.g., minimizing durations of maintenance activities and minimizing work on redundant trains of equipment).

EGC Response to APLA RAI 02, Part a.iv:

The evaluation process described above aligns with NUREG-1855, Revision 1, with respect to Stage E – Assessing Model Uncertainty as described below.

- Step E-1.1 (Identification of Sources of Model Uncertainty and Related Assumptions): Appendix B of EPRI TR 1026511 was used to identify potential sources of model uncertainty in the fire PRA. Unique plant-specific issues were also considered in the identification process for the PRA models.

- Step E-1.2 (Identification of Relevant Sources of Model Uncertainty and Related Assumptions): This step allows for screening of potential sources of model uncertainty based on the parts of the models used for the application. Since the Risk-Informed Completion Time evaluations involve complete model re-quantification for each case analyzed, no specific potential sources of uncertainty were screened out from further consideration for the application.

- Step E-1.3 (Characterization of Sources of Model Uncertainty and Related Assumptions): Per the guidance in NUREG-1855 and the associated EPRI reports, the characterization process involves identifying: 1) the part of the PRA model affected; 2) the modeling approach or assumptions utilized in the model; 3) the impact on the PRA model; and 4) representation of conservative bias (if applicable). These considerations were included in the evaluation of the potential sources of model uncertainty for Byron and Braidwood.

- Step E-1.4 (Qualitative Screening of Sources of Model Uncertainty and Related Assumptions): This step allows for screening out potential sources of model uncertainty by referencing consensus model approaches. The evaluation process for Byron and Braidwood included identifying the approach utilized (e.g., consensus approach or other applicable guidance) and using those considerations as the means to qualitatively screen potential impacts on the application.

- Step E-1.5 (Identification and Characterization of Relevant Sources of Model Uncertainty and Related Assumptions Associated with Model Changes): The implementation of the RICT program utilizes the base PRA models with some specific optimization to improve quantification speed. The FPIE PRA model logic structure is not structurally altered for optimization. For the Fire PRA model, where there is a group of fire initiators that have the exact same event impact, the group of initiators may be quantified using a single representative initiator for the group. In these cases, the representative event frequency is set to the maximum initiating event frequency for the group for the quantification, and then reset to the sum of the group frequency to ensure that the CDF/LERF value for the configuration, and therefore the RICT calculation, is not under-estimated for the truncation value utilized. The results for optimized portions of the Real Time Risk model will be reviewed to ensure a logically equivalent result is achieved in comparison to the Model of Record. As such, no new sources of model uncertainty have been introduced for the application.

- Step E-2 (Identification of Key Sources of Model Uncertainty and Related Assumptions): As described in NUREG-1855, only the relevant sources of uncertainties and related assumptions with the potential to challenge the application's acceptance guidelines are considered key. Also, per NUREG-1855, if any sources of uncertainty do challenge the acceptance guidelines, then appropriate compensatory measures or performance monitoring should be identified to help minimize the risk. In the case of RICT, appropriate compensatory measures will be in place prior to the RMAT being exceeded and for the remaining duration of the RICT configuration. RMAs will be developed as described in Enclosure 12 of the LAR using insights from the PRA models and other good practices (e.g., minimizing durations of maintenance activities and minimizing work on redundant trains of equipment). Additionally, Enclosure 11 of the LAR describes the performance monitoring that will be associated with the RICT program at Byron and Braidwood. As such, the overall RICT program implementation is consistent with Step E-2 of NUREG-1855.

EGC Response to APLA RAI 02, Part b:

No additional plant specific or generic fire PRA key assumptions or uncertainties were identified. Per the process described in NUREG-1855, Revision 1 and EPRI TR 1026511, the key assumptions and uncertainties were summarized in Enclosure 9 of the LAR. As described in Enclosure 12 of the LAR, RMAs will be developed, when appropriate, using insights from the PRA model results specific to the configuration. This will help to minimize the risk and offset any potential issues associated with specific sources of uncertainty for the configuration.

EGC Response to APLA RAI 02, Part c:

No additional plant specific or generic key assumptions or uncertainties were identified. The disposition of most of the candidate sources as not being key involved use of consensus or generally accepted approaches or noting that a slight conservative bias was used in model development. Although conservative bias approaches may lead to masking delta risk increases in some cases, these approaches should generally not contribute significantly to the base risk values. In a bounding case, the calculated delta risk that is potentially masked would be no more than the base case value (i.e., it is zero if not masked and is the base value when the potentially masked component is taken out of service). Therefore, the calculated changes to RICT would be minimal as shown in the illustrative example below which assumes a total masked contribution of 5%.

- Assume total base CDF from all contributors is 1.0E-5.
- Assume masked contribution is up to 5% of that total: 0.05 * 1.0E-5 = 5.0E-7
- For various RICT times the calculated required increase in CDF (Delta-CDF) (i.e., 1E-5/(# Days/365)) and potential impact from masking (at 5% of base CDF) is shown below.

Calculated RICT	Delta-CDF	Adjusted RICT (with 5E-7 masked delta risk)	Ratio to Default Calculation
5.0 Days	7.30E-4	4.997 Days	99.9%
10.0 Days	3.65E-4	9.986 Days	99.9%
20.0 Days	1.83E-4	19.95 Days	99.7%
30.0 Days	1.22E-4	29.88 Days	99.6%

EGC Response to APLA RAI 02, Part d:

The point estimates (CDF and LERF) for the internal events/internal flood and fire PRAs have been compared to the corresponding mean values considering SOKC. For the fire PRA a parametric uncertainty evaluation was performed for the most recent fire PRA model supporting this application. For the FPIE PRA this was done for a recent previous model for which the results are sufficiently close to the current model results that the conclusion is judged to remain valid. The maximum delta risk for any of the units between the point estimate values and mean risk values for FPIE/IF and FPRA, for CDF and LERF based on these parametric uncertainty evaluations is provided below:

	FPIE/IF (per reactor year)	FPRA (per reactor year)
Delta CDF	2E-07	1E-07
Delta LERF	2E-07	5E-08

When compared to the total risk numbers provided in the response to APLA RAI 10, these small deltas do not alter the conclusion with respect to the risk results being within the guidance of RG 1.174.

APLA RAI 03 - Specific Key Assumptions and Sources of Uncertainty

The NRC SE of NEI 06-09, revision 0-A states:

> "When key assumptions introduce a source of uncertainty to the risk calculations (identified in accordance with the requirements of the ASME standard), TR NEI 06-09, Revision 0, requires analysis of the assumptions and accounting for their impact to the RMTS calculated RICTs."

Exelon's letter dated December 13, 2018, Enclosure 9, Table E9-1 and Table E9-3, provide an assessment of key assumptions and sources of modeling uncertainty for the internal events and fire PRAs, respectively.

The disposition regarding the uncertainty associated with cable routing for the fire PRA states "PRA credited components for which cable routing information was not provided represent a source of uncertainty (conservatism) in that these components are assumed failed unnecessarily." It is not clear from the disposition how many or which systems were assumed failed because of the lack of cable routing information and what impact this assumption may have on the RICT calculations. Though the assumption used in the fire PRA model is conservative, NRC staff notes that conservatism in PRA modeling could have a non-conservative impact on the RICT calculations. If an SSC is part of system not credited in the fire PRA or it is supported by a system that is assumed to always fail, then the risk increases due to taking that SSC out of service could be masked by the conservative modeling. Therefore, address the following for the Byron and Braidwood Stations:

a. Identify the systems or components that are assumed to be always failed in the PRA and justify that this assumption has an inconsequential impact on the RICT calculations.

b. As an alternative to item a, above, propose a mechanism to ensure that a sensitivity study is performed for the RICT calculations for applicable SSCs to determine the impact of the conservative modeling on the RICT. The proposed mechanism should also ensure that any additional risk associated with the modeling is either accounted for in the RICT calculation or is compensated for by applying an additional RMA during the RICT.

EGC Response to APLA RAI 03, Part a:

The components that are assumed to be always failed in the fire PRA due to unknown cable routing are known as the Unknown Location (UNL) components. The systems that contain UNL components are listed in Table 1. The UNL components in these systems were examined for impact to the RICT program.

Table 1 provides a discussion of impacts of the components treated as UNL. For most of the systems, the UNL components do not fall within the scope of the RICT program and do not serve support functions for RICT LCOs. However, a small number of UNL components within several systems require further evaluation as noted in the response to APLA RAI-03, Part b.

Table 1. Systems with Components Assumed Always Failed in the Fire PRA

System ID	System Description	UNL Items in RICT Scope?	UNL Components Modeled as Supporting RICT Functions?	Disposition	Further Evaluate RICT Impact Prior to Implementation?
CI	Containment Isolation	Yes	Yes	UNL Components include Manual Phase A & B isolation switches.	Yes
CS	Containment Spray	Yes	Yes	UNL Components include CS Pumps and associated valves.	Yes
RC	Reactor Coolant System	Yes	Yes	UNL components are temperature switches, flow transmitters, delta T indicators, and process logic modules. These components are modeled in the PRAs to support the automatic reactor trip functions within the RICT program scope. However, consistent with standard practice, the fire PRA assumes a successful reactor trip at fire transient onset, and screens fire-induced reactor trip failures due to diverse and redundant reactor trip capability. Therefore, the assumed failure of these UNL components does not lead to masking of FPRA results or affect the FPRA deltas for RICT.	No
RY	Pressurizer	Yes	Yes	UNL components are low-pressurizer pressure logic indicators which feed into reactor trip inputs. These components are modeled in the PRAs to support the automatic reactor trip functions within the RICT program scope. However, consistent with standard practice, the fire PRA assumes a successful reactor trip at fire transient onset, and screens fire-induced reactor trip failures due to diverse and redundant reactor trip capability. Therefore, the assumed failure of these UNL components does not lead to masking of FPRA results or affect the FPRA deltas for RICT.	No
AR	Area Radiation Monitoring	No	Yes	AR is not in scope of the RICT program. However, AR monitors provide input for manual Phase A Isolation.	Yes
DO	Diesel Fuel Oil	No	Yes	UNL components are day tank level switches. Redundant trains and manual actions are modeled in the PRA to support refilling of diesel fuel oil tanks.	Yes
IA	Instrument Air	No	Yes	UNL components are the inbound/outbound containment IA valves.	Yes

System ID	System Description	UNL Items in RICT Scope?	UNL Components Modeled as Supporting RICT Functions?	Disposition	Further Evaluate RICT Impact Prior to Implementation?
SA	Station Air	No	Yes	UNL components are service air compressors. SA supports Instrument Air.	Yes
VA	Ventilation	No	Yes	UNL components support charcoal filtration and ventilation operations as well as cubicle cooling for the CS pumps.	Yes
AP	Auxiliary Power (AC)	No	No	UNL components support functions outside of RICT scope. PRA modeled components are non-safety related and modeled to support other non-safety-related systems and functions outside of RICT scope, including CB/CD, WS, CW, and FW.	No
CD	Condensate / Condensate Booster	No	No	CD is not in scope of the RICT program and does not directly support other RICT functions.	No
CV	CVCS/Charging	No	No	UNL Components are primarily related to pressurizer aux spray and boration via the boric acid storage tank. These are not functions covered within the RICT program for CV, and therefore, the UNL components are outside of the RICT program scope.	No
CW	Circulating Water	No	No	CW is not in scope of the RICT program and does not directly support other RICT functions.	No
DC	DC Power	No	No	UNL components support functions including Reactor Containment Fan Cooler operation, CV regeneration modes, and CW makeup (Byron only). These functions are not in the scope of the RICT program.	No
FP	Fire Protection	No	No	UNL components support functions outside of RICT program scope. The motor driven fire pump (train A) is the UNL component. It is primarily modeled in the PRA to provide backup cooling to the CV pump oil cooler upon loss of SX. The Diesel-driven fire pump (train B) provides functional redundancy. Additionally, this backup function is only required in cases where a complete loss of essential service water occurs. Since the model accounts for redundancy for this backup support function, any impact to RICT would be small.	No
FW	Main Feedwater	No	No	FW is not in scope of the RICT program. Therefore, there is no impact from UNL treatment.	No

System ID	System Description	UNL Items in RICT Scope?	UNL Components Modeled as Supporting RICT Functions?	Disposition	Further Evaluate RICT Impact Prior to Implementation?
MS	Main Steam	No	No	UNL components are the Steam Dump Valves, which are outside of the RICT program scope. The function failed is associated with the opening of the SDVs, not closure of SDVs as required to support a post-fire shutdown. SDVs do not directly support other RICT functions.	No
VX	MEER Room Ventilation and Switchgear Room Heat Removal	No	No	UNL components are HELB dampers for the switchgear rooms. HELB is not a consideration of the Fire PRA. There is no impact to RICT calculations.	No
WS	Non-Essential Service Water	No	No	WS is not in scope of the RICT program and does not directly support other RICT functions.	No
WW (BY)	Deep Well Makeup System	No	No	WW is not in scope of the RICT program and does not directly support other RICT functions.	No

EGC Response to APLA RAI 03, Part b:

As identified in the response to part a above, additional evaluation is needed for a small set of UNL components to demonstrate that RICT calculations are not significantly affected by the conservatively-assumed failure in the base fire PRA quantification, or to establish appropriate risk management actions to compensate for these impacts. To ensure the calculated RICTs for these functions are not significantly affected by the UNL assumptions, sensitivity studies will be performed prior to implementation of RICT to demonstrate minimal impact to the calculated RICT values. Where this cannot be demonstrated, RMAs will be developed for the affected RICT LCOs.