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10 CFR 50.90

August 27, 2019

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

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Subject: Response to Request for Additional Information
License Amendment Request to Revise 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 J. Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, “License Amendment Request to Revise 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. ML18347B366).

2. Letter from D. Helker (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, “Supplement to License Amendment Request to Revise 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. ML19045A011).

3. Electronic mail message from V. Sreenivas, U.S. Nuclear Regulatory Commission, to G. Stewart, Exelon Generation Company, LLC, "Limerick-Request for Additional Information: Risk Informed Completion Times TSTF-505, Revision 2, ‘Provide Risk-Informed Completion Times – RITSTF Initiative 4b’ (EPID L-2018-LLA-0567)," dated July 10, 2019 (ADAMS Accession No. ML19192A031).

4. Letter from D. Gudger (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 2, ‘Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b’," dated August 12, 2019 (ADAMS Accession No. ML19224B705).

By letter dated December 13, 2018 (Reference 1), as supplemented by letter dated February 14, 2019 (Reference 2), Exelon Generation Company, LLC (Exelon) requested an amendment to the Renewed Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (Limerick), Units 1 and 2, respectively.

The proposed amendment 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” (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18183A493).

During June 17, 2019 to June 21, 2019 the U.S. Nuclear Regulatory Commission (NRC) staff conducted an audit at Exelon’s offices in Kennett Square, Pennsylvania to support development of its safety evaluation. Upon completion of the audit, the NRC staff determined that additional information is needed to complete its review of the LAR.

A draft request for additional information (RAI) was provided to G. Stewart (Exelon) by electronic email dated July 2, 2019. A conference call was subsequently held with the NRC on July 10, 2019 to provide clarification of the draft RAI questions. The formal RAI was issued by electronic email to G. Stewart (Exelon) on July 10, 2019 (Reference 3).

As noted in Reference 3, due to the need to perform sensitivity analyses and other additional studies, response to APLA RAI-02, RAI-03 and RAI-08 is required by August 30, 2019. The response to all other RAIs was submitted to the NRC by Reference 4.

The attachment to this letter provides a restatement of the NRC questions APLA RAI-02, RAI-03 and RAI-08 followed by our responses.

Exelon has reviewed the information supporting a finding of no significant hazards consideration, and the environmental consideration, that were previously provided to the NRC in Attachment 1 of the Reference 1 letter. Exelon has concluded that the information provided in this response does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92. In addition, Exelon has concluded that the information in this response does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), Exelon is notifying the Commonwealth of Pennsylvania of this supplement to the application for license amendment by transmitting a copy of this letter and its attachment to the designated State Official.

This letter contains no regulatory commitments.

If you should have any questions regarding this submittal, please contact Glenn Stewart at 610-765-5529.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 27th day of August 2019.

Respectfully,

David P. Helker
Sr. Manager - Licensing
Exelon Generation Company, LLC

Attachment: License Amendment Request - Response to Request for Additional Information

cc: USNRC Region I, Regional Administrator
USNRC Project Manager, Limerick
USNRC Senior Resident Inspector, Limerick
Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Protection

ATTACHMENT 1

License Amendment Request

Limerick Generating Station, Units 1 and 2
NRC Docket Nos. 50-352 and 50-353

Response to Request for Additional Information
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A restatement of the NRC questions APLA RAI-02, RAI-03 and RAI-08 followed by our responses is provided below.

A. <u>PROBABILISTIC RISK ANALYSIS (PRA) LICENSING BRANCH A (APLA)</u>

<u>APLA RAI-02 - Specific Key Assumptions and Sources of Uncertainty</u>

The NRC SE for NEI 06-09 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 [risk-managed technical specifications] calculated RICTs."

a. Regarding the uncertainty associated with continued injection from control rod drive (CRD) after containment failure, the disposition in LAR Table E9-1 states that RMAs will be implemented to address the uncertainty with this assumption for RICTs that are pertinent to loss of containment heat removal scenarios. Address the following:

    i. Identify the Limiting Conditions for Operation (LCOs) proposed to be included in the RICT program for which loss of containment heat removal scenarios affect the RICT.

    <u>Response</u>

    The Limiting Conditions for Operation (LCOs) proposed to be included in the

Limerick RICT program for which the containment heat removal function is directly impacted are the following:

TS/LCO Condition	
3.6.2.2.a	One suppression pool spray loop inoperable
3.6.2.3.a	One suppression pool cooling loop inoperable
3.7.1.1.a.2	One RHRSW pump in each subsystem inoperable
3.7.1.1.a.3	One RHRSW subsystem inoperable
3.7.1.1.a.3.a	One RHRSW subsystem inoperable
3.7.1.1.a.3.b	One RHRSW subsystem inoperable
3.7.1.2.a.3	One emergency service water loop inoperable

Note that this list excludes LCOs 3.7.1.1.a.6 and 3.7.1.2.a.4 which are excluded from the RICT program scope in accordance with Reference 4.

ii. Provide the results of a sensitivity study of the impact on RICTs of this assumption to credit control rod drive injection after containment failure. Discuss the results of this sensitivity study in the context of the RICT estimates provided in Table E1-2 of Enclosure 1 of the LAR.

Response

Sensitivity cases were performed for the set of TS/LCOs listed above assuming that CRD fails after containment failure. This is a very bounding assessment because the most likely containment failure mode under overpressure situations is a leak in the upper drywell region, which would not render CRD inoperable after containment failure. The results of the bounding sensitivity cases with respect to the calculated RICT times reported in Table E1-2 of the Limerick TSTF-505 LAR are shown below. The results of the bounding sensitivity cases generally show a nominal increase in CDF values, but the calculated values remain above the 30-day backstop. In these cases, the risk increase is almost entirely due to scenarios with operator action failures to vent containment given a loss of suppression pool cooling. Accounting for the RMA of briefing plant Operators on the importance of this action would greatly reduce the calculated risk increase from the bounding sensitivity case result. Also note that reasonable variations of this source of uncertainty (i.e., 2x or 5x the current value) would have much less of an impact on the CDF increase than does the bounding sensitivity case. As presented here, for the proposed LCOs, the sensitivity performed demonstrates that there is no impact on the RICT estimates shown in Table E1-2 of the LAR.

TS/LCO Condition	Original RICT Estimate (Days)	Sensitivity Case RICT Estimate (Days)
3.6.2.2.a One suppression pool spray loop inoperable	30.0	30.0

TS/LCO Condition	Original RICT Estimate (Days)	Sensitivity Case RICT Estimate (Days)
3.6.2.3.a One suppression pool cooling loop inoperable	30.0	30.0
3.7.1.1.a.2 One RHRSW pump in each subsystem inoperable	30.0	30.0
3.7.1.1.a.3 One RHRSW subsystem inoperable	30.0	30.0
3.7.1.1.a.3.a One RHRSW subsystem inoperable	30.0	30.0
3.7.1.1.a.3.b One RHRSW subsystem inoperable	30.0	30.0
3.7.1.2.a.3 One emergency service water loop inoperable	30.0	30.0

iii. Describe the RMAs (e.g., operator briefing on the significant human actions in the PRA that are pertinent to loss of containment heat removal scenarios) to be implemented for applicable RICTs and provide justification that these RMAs minimize the potential adverse impact on the RICT.

Response

Reliance on CRD after containment failure arises when all modes of containment heat removal fail. The significant human actions that would avoid that situation are the actions to align suppression pool cooling and the actions to vent containment at the Primary Containment Pressure Limit (PCPL) should suppression pool cooling fail. RMAs associated with pre-job briefs for the importance of these actions will help reduce the likelihood of accident progression to containment failure and subsequently relying on CRD for continued injection. Beyond that, the importance of the potential need to refill the CST to maintain injection after containment failure or containment venting is the other operator action that can be included in the RMAs for these TS/LCOs. Note that, independent of the sensitivity study results presented here, these same actions would be identified as important from the base analysis for the configuration as well.

b. LAR Table E9-1 identifies that, given an uncontrolled flooding of the steam lines, a nominal failure probability of 1×10^{-3} is assigned to Safety Relief Valves (SRVs) being permanently disabled, which precludes the ability to depressurize the reactor pressure vessel through the SRVs. The disposition states that the 1×10^{-3} failure probability provides a slight conservative bias to the results such that the impact on RICT calculations is not unduly influenced. The LAR identifies that this uncertainty affects the RICTs for the LCOs associated with the High Pressure Injection Systems. The disposition further identifies that although the SRVs are designed to pass water, they are never tested in this fashion.

Discuss the sensitivity of this assumption for the RICT application. If determined to be significant to the RICT application, discuss the RMAs that will be implemented to minimize the impact of this assumption.

Response

The PRA models an unlikely scenario in which steam lines may become flooded if all of the following occur:

- level is not maintained below Level 8 (the high-high level setpoint),
- the automatic HPCI, RCIC and FW system trip functions fail, and
- operators do not respond in time to take manual control of HPCI or RCIC after the Level 8 high-high level trip failure.

The water carried over into the steam lines is then postulated as having the potential to disable the SRVs from being able to subsequently perform their overpressure control function (i.e., fail to open to relieve pressure) even if the RPV water level drops later.

The above scenario, i.e., failure of the HPCI, RCIC or FW systems to trip on high reactor level, is included in the PRA model as leading to a steam line flooding scenario. Given the conditions occur that would allow initial flooding of the steam lines, a probability is assigned that this flooding permanently disables all of the SRVs. This is modeled with basic event APHSRXDXI(2), which is basically a common cause failure to open event for the SRVs.

Per the Limerick PRA Data Notebook:

> *The passing of water through the SRVs temporarily (i.e. should Level 8 trips fail and RPV water level rises to the main steam lines) should not render the SRVs totally unavailable. However, to account for this possibility, a failure probability of 1.0E-3 is assigned to this occurring (APHSRXDXI(2)) given that conditions arise to allow water ingress into the main steam lines.*

Thus, the model assumes a failure probability of 1.0E-3 for the common cause failure to open of the SRV pressure relief function for the scenario described above, i.e., following flooding of the steam lines. Note that since this is a scenario in which relief through the SRVs is required, potential damage to the SRVs due to initial water relief would be a beneficial failure, i.e., a likely failure mechanism due to water relief through SRVs is damage to the valve seat resulting in leakage through the valve. Such leakage would be a beneficial failure that is not credited in the model.

For non-steamline flooding scenarios in the Limerick PRA, controlled reactor depressurization will be successful if at least two of 14 safety/relief valves open and remain open. For these scenarios, there is a Common Cause Failure (CCF) basic event (ARVALLCPI) modeled for 13 of 14 SRVs failing to open which has a probability value of 1.9E-6. This is the same common cause failure as APHSRXDXI(2) but in other scenarios in which there is no initial water relief challenge.

Thus, modeling all SRVs as unavailable for the scenario with water in the steam lines with a 1.0E-03 failure probability imposes an increase by more than a factor of 500 compared to the CCF failure to open basic event ARVALLCPI probability. This is considered to be a reasonable representation given the large number of SRVs relative to the number required, and likely represents a slight conservative bias.

In the base CDF model, the relative FV importance of event APHSRXDXI is 3.2E-4. Use of a probability of 1.0E-02 was examined as a sensitivity. With the 1.0E-02 common cause failure probability, i.e., a factor of 5000 greater than the nominal CCF, the relative importance of event APHSRXDXI is 5.6E-3. This demonstrates that the model is not sensitive to the existing conservative value.

Additionally, a similar sensitivity case (i.e., increase APHSRXDXI to 1.0E-2) was performed for the specific configurations of HPCI and RCIC out of service. The results of these sensitivity cases with respect to the calculated RICT times reported in Table E1-2 are shown below. The results of the sensitivity cases generally show a nominal increase in CDF values, but the calculated RICT values remain above the 30-day backstop. In these cases, the risk increase is almost entirely due to scenarios with a loss of feedwater and condensate followed by a loss of HPCI and RCIC and operators fail to depressurize. Note that independent of the sensitivity study results presented here, the same RMAs would be identified as important from the base analysis for the configuration as well.

TS/LCO Condition		Original RICT Estimate (Days)	Sensitivity Case RICT Estimate (Days)
3.5.1.c.1	HPCI system inoperable	30.0	30.0
3.7.3.a	RCIC system inoperable	30.0	30.0

APLA RAI-03 - 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 (FLEX) coping strategies and equipment 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 the LAR as possible RMAs, the LAR and other docketed information do not indicate if Limerick has 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 mitigating actions into the Limerick internal events, including internal flooding, or fire PRA models. If not incorporated or their inclusion is not expected to impact the PRA results used in the RICT program, no additional response is requested.

   Response

   Exelon has credited FLEX equipment and mitigating actions in the Limerick internal events, internal flooding, and fire PRA models.

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

   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.

Response

Credit is taken for the following in each of the PRA models:

1) Deploying and aligning the portable FLEX 480V generators (limited to extended loss of AC power (ELAP) scenarios).

2) Deploying and aligning the portable FLEX pumps (limited to ELAP scenarios).

3) Prolonged RCIC operation via partial RPV depressurization and venting containment using the permanently installed Hardened Containment Vent System (HCVS).

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.

Response

The portable FLEX pumps and generators are not like other plant equipment. The reliability data for these components is treated as noted in response to item iii below. The HCVS includes permanently installed components similar to other permanently installed plant components.

The credited operator actions for each of the mitigating strategies listed above are shown below. These actions are similar to other operator actions included in the PRA models and are evaluated using approaches consistent with the endorsed ASME/ANS RA-Sa-2009 PRA standard as documented in the Limerick HRA notebook.

1) Success of the FLEX generators includes required operator actions for DC Load Shed (QHULSFDXI), deploy and start the FLEX generators (QHUFBXDXI), align the FLEX generators to the battery chargers (QHUFB1DXI or QHUFB2DXI), and refuel the FLEX generators (QHUREFDXI).

2) Success of the FLEX pumps includes required operator actions for aligning the FLEX pumps from the fire water system (QHUFFXDXI), aligning the FLEX pumps for RPV injection from the spray pond (QHULPADXI or QHULPBDXI), and refueling the FLEX pumps (QHUPRFDXI).

3) Success of prolonged RCIC operation includes required operator actions for performing partial RPV depressurization (RHUVT1DXI), opening of the hardened vent at the HCVS panel (QHUCVEDXI), aligning the FLEX pumps for suppression pool makeup from the spray pond (QHUSPADXI or QHUSPBDXI), and refueling the FLEX pumps (QHUPRFDXI).

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.

Response

While industry data for FLEX components is being collected and is anticipated to be available prior to implementation of RICT at Limerick, an assumption is made that two-times the generic reliability values for similar equipment provides a reasonable approximation of the reliability of the FLEX equipment. The values used for the reliability data of the FLEX generators and FLEX pumps are shown below.

Failure Mode	FLEX Generators	FLEX Pumps
Fail to Start	1.3E-2	6.3E-3
Fail to Run (24 Hours)	7.3E-2	9.5E-2
Total Failure Probability	**8.6E-2**	**1.0E-1**

The uncertainties associated with the data values are based on the uncertainty parameters from the generic data, and are in accordance with the ASME/ANS PRA Standard.

Use of these values should provide a reasonable approximation of the reliability of the FLEX equipment. This assumption will be revisited when the generic industry data for FLEX equipment becomes available.

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.

   Response

   All the human error probabilities for FLEX components were evaluated with the same methodology used for all human error probabilities in the Limerick PRA models as documented in the Limerick HRA notebook. As such, the HEPs include an evaluation of the scenario specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of the ASME/ANS RA-Sa-2009 PRA Standard.

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.

Response

Maintenance procedures for the portable equipment were not reviewed for possible pre-initiator human failures. Basic events for maintenance unavailability of the pumps and generators are included in the model with screening values of 1E-02. These maintenance terms represent failure of the entire system (i.e., all the FLEX generators for the unit or all the FLEX pumps for the unit) as credit for the N+1 FLEX components are not included in the model. With the screening values, the importance measures for these maintenance terms do not exceed the risk-significance threshold per the PRA standard. This low risk significance of the maintenance events, combined with the fact that the FLEX equipment failure probability values and combined required post-initiator HEPs used in the model far exceed this maintenance unavailability screening value, allows pre-initiator human failures for the FLEX components to be screened from inclusion in the model.

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.

   Response

   All the procedures for use of FLEX equipment or mitigating strategies at Limerick are clear, explicit, and built into the Emergency Operating Procedures (including the requirement for declaration of an ELAP). The declaration of an ELAP is clear and explicit in the Limerick E-1 procedure (LOSS OF ALL AC POWER (STATION BLACKOUT)).

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.

   Response

   As described in the other responses to this RAI, no new methodologies were implemented for the FLEX equipment and mitigating strategies that were added to the PRA model. The impacts of the modeling changes on the results were as expected (i.e., reductions in the

SBO and total loss of AC accident sequences when FLEX generators are credited, and reductions in the loss of containment heat removal scenarios when the HCVS system is credited). These changes do not represent a change in capability that impacts the significant accident sequences, but merely represent model updates to ensure that the models reflect the as-built, as-operated plant.

d. Section 2.3.4 of NEI 06-09, Revision 0-A, states that PRA modeling uncertainties shall be considered in application of the PRA base model results to the RICT program. The NRC 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 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 (e.g., any multipliers for failure probabilities) 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 the 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.

Response

Based on the types of scenarios for which credit for FLEX equipment is modeled as described above (i.e., reductions in the SBO and total loss of AC accident sequences when FLEX generators are credited), the following LCOs were identified as potentially being the most sensitive to assumptions related to the FLEX modeling.

TS/LCO Condition	
3.7.1.2.a.3	One emergency service water loop inoperable

TS/LCO Condition	
3.8.1.1.d	One offsite circuit and one diesel generator inoperable
3.8.1.1.f	One offsite circuit inoperable
3.8.1.1.g	Two offsite circuits inoperable
3.8.2.1.a.3	Two battery chargers on one division inoperable
3.8.2.1.c	Any battery(ies) on one division of required DC electrical power sources inoperable
3.8.3.1.a	One required AC distribution system divisions not energized
3.8.3.1.b	One required DC distribution system divisions not energized

Note that this list excludes LCOs 3.8.1.1.b and 3.8.1.1.h which are excluded from the RICT program scope in accordance with Reference 4.

As described in the response to Item b.iii above, the current assumption is made that two-times the generic reliability values for similar equipment provides a reasonable approximation of the reliability of the FLEX equipment. This leads to probabilities of failure that in total are close to 0.1 or approximately 1-in-10. To further explore the sensitivity to this assumption, the TS/LCO conditions noted above were re-run for RICT using 5x the generic failure probabilities instead of 2x the generic failure probabilities. The increase in these values to a total system failure probability of approximately 1-in-4 represents a conservative bias to the analysis.

The results of the bounding sensitivity cases with respect to the calculated RICT times reported in Table E1-2 are shown below. The results of the sensitivity cases generally show a nominal increase in CDF and LERF values which corresponds to only a small change in the calculated RICT estimates.

TS/LCO Condition		Original RICT Estimate (Days)	Sensitivity Case RICT Estimate (Days)
3.7.1.2.a.3	One emergency service water loop inoperable	30.0	30.0
3.8.1.1.d	One offsite circuit and one diesel generator inoperable	30.0	30.0
3.8.1.1.f	One offsite circuit inoperable	30.0	30.0
3.8.1.1.g	Two offsite circuits inoperable	30.0	28.9
3.8.2.1.a.3	Two battery chargers on one division inoperable	30.0	30.0
3.8.2.1.c	Any battery(ies) on one division of required DC electrical power sources inoperable	16.2	15.5
3.8.3.1.a	One required AC distribution system divisions not energized	7.8	7.5
3.8.3.1.b	One required DC distribution system divisions not energized	16.2	15.5

In the sensitivity cases performed above, the changes to the FLEX reliability values led to shifts in the cutsets which included those reliability values, but this did not result in the identification of any new important operator actions nor did it identify any additional RMAs.

APLA RAI-08 - PRA Modeling of Instrumentation and Controls

The proposed TS limiting conditions for operations (LCOs) include those related to instrumentation and controls (I&C).

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 is 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.

Response

Instrumentation is explicitly modeled in the PRA as required to support the modeled system. The failure data is handled in the same fashion as other components with Type Codes assigned for each component type. Each of the instrumentation Technical Specifications in the scope of the RICT program is listed below with the associated scope of PRA modeling (examples of modeled components and component types). For those functions that are not explicitly modeled, a conservative surrogate event is used to calculate the RICT as noted in the LAR. Additional instrumentation not in the table is similarly modeled to support other systems and operator actions.

RPS INSTRUMENTATION MODELED		
TS TABLE 3.3.1-1 FUNCTION	**EXAMPLE MODELED COMPONENTS**	**COMPONENT TYPE**
1. Intermediate Range Monitors	Not modeled – channel failure used as surrogate	N/A
2. APRM upscale	APRM relays (C71A-K12A)	Relay
	APRMs	APRM
	VOTERs	Voter Circuit
3. RPV high pressure	RPV high pressure relay C71A-K5A	Relay
	PT-042-1N078A	Pressure Transmitter
	PIS-041-1N678A	Trip unit

RPS INSTRUMENTATION MODELED		
TS TABLE 3.3.1-1 FUNCTION	**EXAMPLE MODELED COMPONENTS**	**COMPONENT TYPE**
4. Reactor vessel water level -low, level 3	RPV low level relay (C71A-K6A)	Relay
	LT-042-1N080A	Level Transmitter
	LIS-042-1N680A	Trip unit
5. Main Steam Isolation Valve (MSIV) closure	MSIV Closure relays (C71A-K3A)	Relay
	MSIV Closure bypass relays (C71A-K11A)	Relay
	MSIV inboard and outboard limit switches (ZS-041-122A)	Limit Switch
6. Deleted from TS		
7. Drywell pressure -high	High Drywell Pressure relay (C71A-K4A)	Relay
	PT-042-1N050A	Pressure Transmitter
	PIS-042-1N650A	Trip unit
8. Scram Discharge Volume level- High	Not modeled – channel failure used as surrogate	N/A
9. Turbine Stop Valve Closure	TSV closure relay (C71A-K10A)	Relay
	TSV position switches ZS-001-104A	Pressure Switch
10. Turbine Control Valve Fast Closure	TCV closure relay (C71A-K08A)	Relay
	30%power bypass relay(C71A-K09A)	Relay
	PS-001-102A	Pressure Switch
11. Reactor Mode Switch Shutdown position	Mode switch	Switch
12. Manual Scram	Not modeled – channel failure used as surrogate	N/A
N/A	Scram relays (C71A-K14A)	Relay

ISOLATION ACTUATION INSTRUMENTATION		
TS TABLE 3.3.2-1 FUNCTION	**EXAMPLE MODELED COMPONENTS**	**COMPONENT TYPE**
1. Main Steam Line Isolation		
a. Reactor Water level	LS-042-1N684A	Trip Unit
	LT-042-1N081A	Transmitter
b.-h.	Input to the high pressure break outside containment initiator; signal contribution will be treated as failed for RICT calculation when out of service	N/A
2. RHR Shutdown Cooling Isolation		
a. Reactor Water level	Input to the high pressure break outside containment initiator; signal contribution will be treated as failed for RICT calculation when out of service	N/A
b. Reactor low pressure permissive	See above	
c. Manual initiation	See above	
3. RWCU		
a. High delta flow	Input to the high pressure break outside containment initiator; signal contribution will be treated as failed for RICT calculation when out of service	N/A
b. High area temp.	See above	
c. High delta temp.	See above	
d. SLCS initiation	C41A-K4A	Relay
e. Reactor low level	Input to the high pressure break outside containment initiator; signal contribution will be treated as failed for RICT calculation when out of service	N/A
f. Manual initiation	See above	
4. HPCI Isolation		
a. High steam line delta pressure	Input to the high pressure break outside containment initiator; signal contribution will be treated as failed for RICT calculation when out of service	N/A

ISOLATION ACTUATION INSTRUMENTATION		
TS TABLE 3.3.2-1 FUNCTION	**EXAMPLE MODELED COMPONENTS**	**COMPONENT TYPE**
b. Low steam supply pressure	See above	
c. High turbine exhaust pressure	See above	
d. High room temp.	See above	
e. High room differential temp.	See above	
f. High pipe tunnel temp.	See above	
g. Manual initiation	See above	
h. HPCI steam line delta pressure timer	See above	
5. RCIC		
a. High steam line delta pressure	Input to the high pressure break outside containment initiator; signal contribution will be treated as failed for RICT calculation when out of service	N/A
b. Low steam supply pressure	See above	
c. High turbine exhaust pressure	See above	
d. High room temp.	See above	
e. High room differential temp.	See above	
f. High pipe tunnel temp.	See above	
g. Manual initiation	See above	
6. Primary Containment		
a. Reactor water level	LT-042-1N081A	Transmitter
b. Drywell pressure	PT-042-1N078A	Transmitter
c-g. (Rad monitors or deleted)	Not modeled - generic isolation failure event used as surrogate	N/A
h. High drywell pressure/low reactor pressure	See above	

ISOLATION ACTUATION INSTRUMENTATION		
TS TABLE 3.3.2-1 FUNCTION	**EXAMPLE MODELED COMPONENTS**	**COMPONENT TYPE**
i. PCIG to drywell delta pressure low	See above	
j. Manual initiation	See above	
7. Secondary Containment	Out of Scope	

ECCS INSTRUMENTATION MODELED		
TS TABLE 3.3.3-1 FUNCTION	**EXAMPLE MODELED COMPONENTS***	**COMPONENT TYPE**
N/A	Core Spray initiation logic	Relay Logic
N/A	RHR LPCI initiation logc	Relay Logic
N/A	HPCI intitiation logic	Relay Logic
N/A	ADS initiation logic	Relay Logic
Reactor low level(s)	LT-042-1N091A	Transmitter
	LIS-042-1N691A	Trip Unit
	LT-042-1N091B	Transmitter
	LS-042-1N692B	Trip Unit
High drywell pressure	PT-042-1N094A	Transmitter
	PIS-042-1N694A	Trip Unit
RPV pressure low permissive	E21A-K25A/B	Relay
	E21A-K27A/B	Relay
	PT-042-1N090A	Transmitter
	PIS-042-1N690A	Trip Unit
Injection valve differential pressure low	PDT-051-1N058A	Transmitter
	PDISL-051-1N658A	Trip Unit
Manual	Not modeled – associated operator action used as surrogate	N/A
LPCI, RHR Pump discharge pressure (ADS permissive)	PT-052-1N055A	Transmitter
	PIS-052-1N655A	Trip Unit
	PT-051-1N055A-D	Transmitter
	PIS-051-1N655A	Trip Unit
	PT-051-1N056A	Transmitter
	PIS-051-1N656A	Trip Unit
CST low level	LT-055-1N061B	Transmitter
	LIS-055-1N661B	Trip Unit
	LT-049-1N035A	Transmitter
Suppression pool level high	Not modeled – channel failure used as surrogate	N/A
Reactor high level	LS-042-1N693B	Trip Unit

ECCS INSTRUMENTATION MODELED		
TS TABLE 3.3.3-1 FUNCTION	**EXAMPLE MODELED COMPONENTS***	**COMPONENT TYPE**
ADS timer	B21C-K8A	Timer
4KV bus under voltage-degraded voltage	Not modeled – channel failure used as surrogate	N/A
4KV bus under voltage loss of voltage	E21A-K18A	Relay
	127-115	Relay
	162-11502 (D11)	Time Delay Relay
	162-11509 (D11)	Time Delay Relay
	183-11509 (D11)	Relay

* Note: Parameters and components are shared among ADS, Core Spray, RHR-LPCI and PCI and are therefore listed once by parameter.

RRCS/SLCS INSTRUMENTATION (ATWS RPT) MODELED		
TS TABLE 3.3.4.1-1 FUNCTION	**EXAMPLE MODELED COMPONENTS**	**COMPONENT TYPE**
1. Reactor level	LT-042-1N402A	Level Transmitter
	ATM 2A	Trip Module
2. Reactor pressure	PT-042-1N403A	Pressure Transmitter
N/A	ATM 3A	Trip Module
N/A	Circuit card A1	Logic Device

EOC-RPT INTRUMENTATION MODELED		
TS TABLE 3.3.4.2-1 FUNCTION	**EXAMPLE MODELED COMPONENT**	**COMPONENT TYPE**
N/A	C71A-K50A/B	Relay
1. TSV closure	ZS-001-104A	Limit switch
2. TCV closure	C71A-K10A	Relay
	PS-001-102A	Pressure switch
	C71A-K8A	Relay

RCIC INSTRUMENTATION MODELED		
TS TABLE 3.3.5-1 FUNCTION	**EXAMPLE MODELED COMPONENTS***	**COMPONENT TYPE**
N/A	RCIC LOGIC	Logic
a. Reactor low level	LT-042-1N097A/E	Transmitter
	LIS-042-1N697A/E	Trip Unit
	LIS-042-1N692A/E	Trip Unit
	LIS-042-1N091A/E	Transmitter
b. Reactor high level	LIS-042-1N693A/E	Trip Unit
	LIS-042-1N698A/E	Trip Unit
	LT-042-N097A/E	Transmitter
c. CST level	LIS-049-1N635A/E	Trip Unit
	LT-049-1N035A	Transmitter
d. Manual initiation	Not modeled – associated operator action used as surrogate	N/A

FEEDWATER/ MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION MODELED		
TS TABLE 3.3.9-1 FUNCTION	**EXAMPLE MODELED COMPONENT**	**TYPE**
1. Reactor high level	Logic reactor high level	Logic
	LT-042-1N004A	Level transmitter

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 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 a RICT calculation and that sensitivity studies should be used to develop appropriate compensatory RMAs.

   Regarding digital I&C, 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 systems and 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 or components is included in the RTR model, then address the following:

   i. Provide the results of a sensitivity study demonstrating that the uncertainty associated with modeling the digital I&C systems or components has an inconsequential impact on the LCOs included in the RICT program, or

ii. If the modeling of digital I&C systems or components is determined to be a key source of uncertainty for the application, identify impacted LCOs and describe how sensitivity studies are used to identify RMAs to minimize the potential adverse impacts of this source of uncertainty, consistent with the guidance in Section 2.3.4 of NEI 06-09, Revision 0-A

Response

Limerick has a digital feedwater control system, installed in 2004 (Unit 1) and 2005 (Unit 2). In the feedwater level control logic, a false signal from the redundant reactivity control system could lead to inadvertent termination of automatic feedwater level control. This event is modeled in the PRA by a basic event (FPHRRCDXI) included in the feedwater system level control logic. In the feedwater level control logic, a high reactor water level (level 8) signal trips the feedwater turbines. An event for a false level 8 signal is included in the feedwater system fault tree.

The Limiting Conditions for Operation (LCOs) proposed to be included in the Limerick RICT program for which loss of digital feedwater most affect the RICT are the following:

TS/LCO Condition	
3.5.1.c.1	HPCI system inoperable
3.7.3.a	RCIC system inoperable

Sensitivity cases were performed for the set of TS/LCOs above assuming that the failure rates of each of the basic events associated with digital feedwater were 100 times greater than the base values. For these sensitivities, the probabilities for the following basic events associated with digital feedwater were each increased by a factor of 100.

Basic Event	Description
FAF450HWI	AF450 FAILS TO AUTO CONTROL LEVEL
FAF100HWI	FIELD BUS AF100A AND AF100B FAIL
FPHRRCDXI	FALSE SIGNAL FROM REDUNDANT REACTIV. CONTROL SYSTEM
FPHL8HDXI	FALSE LEVEL 8 SIGNAL

The results of the sensitivity cases with respect to the calculated RICT times reported in Table E1-2 of the Limerick TSTF-505 LAR are shown below. The results of the sensitivity cases generally show a very nominal increase in CDF values but the calculated RICT values decreased by less than 1% and remained above the 30-day backstop in each case. In these cases, the risk increase is almost entirely due to scenarios with operator actions involving failure to vent containment and failure to swap feedwater to manual control given a loss of automatic control. The RMA of briefing Operators on the importance of these actions would greatly reduce the calculated risk increase from the sensitivity case result. As presented here, for the proposed LCOs, the sensitivity performed demonstrates that there is no impact on the RICT estimates shown in Table E1-2 of the LAR.

TS/LCO Condition		Original RICT Estimate (Days)	Sensitivity Case RICT Estimate (Days)
3.5.1.c.1	HPCI system inoperable	30.0	30.0
3.7.3.a	RCIC system inoperable	30.0	30.0

REFERENCES

1. Letter from J. Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, “License Amendment Request to Revise 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. ML18347B366).

2. Letter from D. Helker (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, “Supplement to License Amendment Request to Revise 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. ML19045A011).

3. Electronic mail message from V. Sreenivas, U.S. Nuclear Regulatory Commission, to G. Stewart, Exelon Generation Company, LLC, "Limerick-Request for Additional Information: Risk Informed Completion Times TSTF-505, Revision 2, ‘Provide Risk-Informed Completion Times –RITSTF Initiative 4b’ (EPID L-2018-LLA-0567)," dated July 10, 2019 (ADAMS Accession No. ML19192A031).

4. Letter from D. Gudger (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 2, ‘Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b’," dated August 12, 2019 (ADAMS Accession No. ML19224B705).