UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 21, 2020

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 − REGULATORY VIRTUAL AUDIT PLAN REGARDING LICENSE AMENDMENT REQUEST TO ADOPT TSTF-505, REVISION 2 (EPID L-2020-LLA-0120)

Dear Mr. Hanson:

By letter dated May 29, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007), Exelon Generation Company, LLC submitted a license amendment request to adopt Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, “Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,” dated July 2, 2018 (ADAMS Accession No. ML18183A493), for the Peach Bottom Atomic Power Station, Units 2 and 3.

The proposed amendments would revise technical specification requirements to permit the use of risk-informed completion times for actions to be taken when limiting conditions for operation are not met.

The U.S. Nuclear Regulatory Commission staff will be conducting a virtual audit from November 9, 2020, to November 13, 2020 (excluding November 11, 2020, which is a Federal holiday), with Exelon Generation Company, LLC staff and associated contractors. The regulatory virtual audit plan is enclosed with this letter.

If you have any questions regarding this matter, please contact me at 301-415-2328 or by e-mail to Jennifer.Tobin@nrc.gov.

Sincerely,

/RA/

Jennifer C. Tobin, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-277 and 50-278

Enclosure:
Regulatory Virtual Audit Plan

cc: Listserv

REGULATORY VIRTUAL AUDIT PLAN

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT

TSTF-505, REVISION 2

EXELON GENERATION COMPANY, LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

1.0 BACKGROUND

By application dated May 29, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007), Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom). The amendments would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF [Risk Informed Technical Specification Task Force] Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC) issued a final model safety evaluation approving TSTF-505, Revision 2, on November 21, 2018 (ADAMS Package Accession No. ML18269A041).

2.0 REGULATORY AUDIT BASES

A regulatory audit is a planned license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. The audit is conducted with the intent to gain understanding, to verify information, and to identify information that will require docketing to support the basis of a licensing or regulatory decision. Performing a regulatory audit is expected to assist the NRC staff in efficiently conducting its review of the LAR and to gain insights of the licensee's processes and procedures. Information that the NRC staff relies upon to make the safety determination must be submitted on the docket.

The basis of this audit is the Peach Bottom LAR to revise TS requirements to permit the use of RICTs and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (ADAMS Accession No. ML071700658).

The audit will be performed consistent with NRC Office of Nuclear Reactor Regulation Office Instruction LIC-111, Revision 1, "Regulatory Audits," dated October 31, 2019 (ADAMS Accession No. ML19226A274). An audit was determined to be the most efficient approach toward a timely resolution of issues associated with this LAR review, since the NRC staff will have an opportunity to minimize the potential for multiple rounds of requests for additional information and ensure no unnecessary burden will be imposed by requiring the licensee to address issues that are no longer necessary to make a safety determination.

3.0 PURPOSE AND SCOPE

The purpose of this audit is to identify information that the licensee should submit on the docket for NRC staff to make a safety determination and to gain a better understanding of the following areas related to the LAR:

- calculations, analyses, and bases underlying the LAR;
- approach for developing and implementing the plant's risk-managed TS program;
- extent that the LAR is consistent with TSTF-505, Revision 2; Nuclear Energy Institute (NEI) Topical Report 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document," dated November 6, 2006 (ADAMS Package Accession No. ML122860402); and the NRC's Final Safety Evaluation for NEI 06-09, dated May 17, 2007 (ADAMS Accession No. ML071200238);
- whether the proposed configurations introduce any adverse effects on the ability or capacity of plant equipment to perform its design-basis function(s) when the plant is operated in the proposed TS allowable configuration;
- technical acceptability of the probabilistic risk assessment (PRA) for use in the application and how plant design features are modeled in the PRA used to support the LAR; and
- use of the Configuration Risk Management Program tool (i.e., PARAGON) to support RICT program implementation.

The areas of focus for the regulatory audit are the information contained in the LAR, the audit information needs listed in the following section of this audit plan, and all associated and relevant supporting documentation (e.g., methodology, process information, calculations, etc.). The relevant supporting documents are identified below.

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The following documentation should be available to the audit team:

1. the documentation specified in Section 4 of the portal audit plan dated August 4, 2020 (ADAMS Accession No. ML20217L346),

2. PRA notebook regarding component data calculations that address SSC mission times, including the emergency diesel generator split mission times,

3. calculation notebook regarding the tornado missile hazard risk value determinations, and

4. any additional supporting documentation that the licensee may determine is responsive to the NRC staff's above information requests.

5.0 AUDIT TEAM

The members of the audit team are anticipated to be:

- Jennifer Tobin, Project Manager, NRC/DORL (Jennifer.Tobin@nrc.gov)
- Todd Hilsmeier, Team Leader, NRC/APLA (Todd.Hilsmeier@nrc.gov)
- Jeff Circle, NRC/APLA (Jeff.Circle@nrc.gov)
- Robert Pascarelli, Branch Chief, NRC/APLA (Robert.Pascarelli@nrc.gov)
- Milton Valentin-Olmeda, NRC/APLC (Milton.Valentin-Olmeda@nrc.gov)
- Wesley Wu, NRC/APLC (De.Wu@nrc.gov)
- Robert Vettori, NRC/APLB, (Robert.Vettori@nrc.gov)
- Bernard Grenier, NRC/APLB (Bernard.Grenier@nrc.gov)
- Henry Marchlewski, NRC/APLB (Henry.Marchlewski@nrc.gov)
- Khoi Nguyen, NRC/EEOB (Khoi.Nguyen@nrc.gov)
- Stephen Wyman, NRC/EEOB (Stephen.Wyman@nrc.gov)
- Ming Li, NRC/EICB (Ming.Li@nrc.gov)
- Norbert Carte, NRC/EICB (Norbert.Carte@nrc.gov)
- Andrea Russell, NRC/STSB (Andrea.Russell@nrc.gov)
- Vic Cusumano, NRC/STSB (Victor.Cusumano@nrc.gov)
- Derek Scully, NRC/SCPB (Derek.Scully@nrc.gov)
- Nageswara Karipineni, NRC/SCPB (Nageswara.Karipineni@nrc.gov)
- Gurjendra Bedi, NRC/EMIB (Gurjendra.Bedi@nrc.gov)
- Yuken Wong, NRC/EMIB (Yuken.Wong@nrc.gov)
- Shie-Jeng Peng, NRC/SNSB (Shie-Jeng.Peng@nrc.gov)
- Mark Wilk, NRC Contractor, Pacific Northwest National Laboratory (mark.wilk@pnnl.gov)

6.0 LOGISTICS

The audit will be conducted remotely from November 9, 2020, to November 13, 2020 (excluding November 11, 2020), between 8:30 a.m. and 4:00 p.m. each day. An entrance briefing will be held at the beginning of the audit, and an exit briefing will be held at the end of the audit. Attachment A of the audit plan provides the proposed agenda for the remote audit. Attachment B contains the audit questions that the NRC staff would like to have prepared dialogue. The NRC project manager will coordinate with the licensee any identified changes to the audit schedule and logistics.

7.0 SPECIAL REQUESTS

The NRC staff would like access to the documents listed in Section 4.0 above through an online portal that allows the NRC staff and contractors to access documents via the internet. The following conditions associated with the online portal must be maintained throughout the duration that the NRC staff and contractors have access to the online portal:

- The online portal will be password-protected, and separate passwords will be assigned to the NRC staff and contractors who are participating in the audit.

- The online portal will be sufficiently secure to prevent the NRC staff and contractors from printing, saving, downloading, or collecting any information on the online portal.

- Conditions of use of the online portal will be displayed on the login screen and will require acknowledgement by each user.

- Username and password information should be provided directly to the NRC staff and contractors. The NRC project manager will provide Exelon the names and contact information of the NRC staff and contractors who will be participating in the audit. All other communications should be coordinated through the NRC project manager.

8.0 DELIVERABLES

An audit summary, which may be public, will be prepared within 90 days of the completion of the audit. If the NRC staff identifies information during the audit that is needed to support its regulatory decision, the staff will issue requests for additional information to the licensee after the audit.

ATTACHMENT A

Proposed Audit Agenda (Revision 0)

Peach Bottom Atomic Power Station, Units 2 and 3,

License Amendment Request to Adopt TSTF-505, Revision 2

Day 1 – Monday, November 9, 2020 (8:30 am to 4:00 pm)*

- Entrance briefing
    - Opening comments by NRC and Exelon Generation Company, LLC (Exelon)
    - Introductions and logistics
- Real-time risk (RTR) model demonstration by Exelon
- Discuss RTR model and calculation of RICT estimates
    - RTR model (including benchmarking, updating, and how seasonal variations are accounted) **(APLA Questions 07, 08, and 09)**
    - PRA functional determination and RICT estimates
    - Treatment of common cause failures for planned and emergent conditions
    - Identification of risk-management actions **(EEOB Question 05)**
- Discuss Key Principle 5, Maintenance Rule and monitoring **(APLA Question 11)**
- Summary of the day[1]
- NRC staff internal meeting

Day 2 – Tuesday, November 10, 2020 (8:30 am to 4:00 pm)*

- Summary of previous day and review open items
- Discuss internal events PRA technical acceptability
    - I&C diversity and modeling in PRA (**EICB Question 01; APLA Question 06)**
    - EDGs, RCIC, HPSW, and vacuum breakers **(APLA Questions 01 to 04)**
    - Credit for FLEX equipment and actions **(APLA Question 05)**

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[1] If discussion topics are completed early, additional discussions for Day 1 may include seismic hazard from Day 4 and/or design-success criteria from Day 4.

- Discuss key assumptions and uncertainties - process **(APLA Question 10)**
    - Summary of the day[2]
    - NRC staff internal meeting

<u>Wednesday, November 11, 2020</u> – Veterans Day observed (no audit)

<u>Day 3 – Thursday, November 12, 2020 (8:30 am to 4:00 pm)*</u>

- Summary of previous day and review open items
- Discuss fire PRA technical acceptability **(APLB Questions 01 to 12)**
    - Summary of the day
    - NRC staff internal meeting

<u>Day 4 – Friday, November 13, 2020 (8:30 am to 4:00 pm)*</u>

- Summary of previous day and review open items
- Discuss seismic hazard **(APLC Questions 01 to 03)**
- Discuss design-success criteria **(STSB Questions 01 and 02; EEEB Questions 01 to 04)**
- Follow-up on any remaining open items
- Summary of audit and exit meeting (tentatively scheduled for 3:30 pm)

* Lunch will be tentatively scheduled from 12:00 pm – 1:00 pm

<u>Acronyms</u>:

APLA	NRC/NRR/PRA Licensing Branch A
APLB	NRC/NRR/PRA Licensing Branch B
APLC	NRC/NRR/PRA Licensing Branch C
EDG	Emergency Diesel Generator
EEEB	NRC/NRR/Electrical Engineering Branch
EICB	NRC/NRR/Instrumentation & Controls Branch
FLEX	Flexible Mitigation Strategies
HPSW	High-Pressure Service Water
I&C	Instrumentation and Control
NRC	U.S. Nuclear Regulatory Commission

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2 If discussion topics are completed early, additional discussions for Day 2 may include seismic hazard from Day 4, design-success criteria from Day 4, and/or fire PRA from Day 3, if not discussed earlier.

NRR	Office of Nuclear Reactor Regulation
PRA	Probabilistic Risk Assessment
RCIC	Reactor Core Isolation Cooling System
RICT	Risk-informed Completion Time
RTR	Real-time Risk
STSB	NRC/NRR/Technical Specifications Branch

ATTACHMENT B

AUDIT QUESTIONS

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT

TSTF-505, REVISION 2

EXELON GENERATION COMPANY, LLC

PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

By application dated May 29, 2020, Exelon Generation Company, LLC (the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (Peach Bottom) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20150A007). The amendment would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," dated July 2, 2018 (ADAMS Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC) issued a final model safety evaluation (SE) approving TSTF 505, Revision 2, on November 21, 2018 (ADAMS Accession No. ML18269A041). The NRC staff has determined that the following information is needed in order to complete its review.

Probabilistic Risk Assessment Licensing Branch A (APLA) Audit Questions

Regulatory Guide (RG) 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML17317A256), states that the scope, level of detail, and technical adequacy of the probabilistic risk assessment (PRA) are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process. The NRC's SE for Nuclear Energy Institute (NEI) Topical Report NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines, Industry Guidance Document," dated November 6, 2006 (ADAMS Package Accession No. ML122860402) (hereafter NEI 06-09), and the NRC's Final Safety Evaluation for NEI 06-09, dated May 17, 2007 (ADAMS Accession No. ML071200238), state that the PRA models should conform to the guidance in RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." The current version is RG 1.200, Revision 2 (ADAMS Accession No. ML090410014), which clarifies the current applicable American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard is ASME/ANS 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 RG 1.200, the quality of the PRA must be compatible with the safety implications of the proposed TS change and the role the PRA plays in justifying the change. RG 1.200 describes a peer review process using ASME/ANS RA-Sa-2009 as one acceptable approach for determining the technical acceptability of the PRA. The primary results of a peer review are the facts and observations (F&Os) recorded by the peer review team and the subsequent resolution of these F&Os. A process to close finding-level F&Os is documented in Appendix X to the NEI guidance documents NEI 05-04, NEI 07-12, and NEI 12-13, titled "NEI 05-04/07-12/12-[13] Appendix X: Close-out of Facts and Observations (F&Os)" (ADAMS Package Accession No. ML17086A431), which was accepted by the NRC in a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427). NEI 06-09 states that the PRA shall meet Capability Category (CC)-II for the supporting requirements of the PRA standard, and any deviations from these capability categories relative to the RMTS program shall be justified.

APLA QUESTION 01 – Probabilistic Risk Assessment Modeling of Emergency Diesel Generators

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that the analysis in Section 4.5.2 of PRA Notebook PB-PRA-013 documented the impact of using a "split" mission time of 4 and 8.2 hours for the emergency diesel generators (EDGs). The results of a sensitivity study in Section 4.5.2, which used a PRA model mission time of 24 hours for the EDGs, demonstrated a 3 percent increase in overall core damage frequency (CDF). It is unclear to the NRC staff how the EDGs would only be required for a specific portion of the PRA analysis window of 24 hours. The NRC staff notes that this source of uncertainty does not appear to have been addressed in PRA Notebook PB-MISC-043, which addresses the impact of PRA assumptions on RICT calculations, especially conditions related to alternating current (AC) and direct current (DC) power (e.g., TSs 3.8.1 and 3.8.4). In light of these observations, provide the following information:

a) Provide justification for the use of split mission times for the EDGs in the Peach Bottom PRA models. Include in this discussion the reasoning for not using the standard 24-hour mission time used in PRA models.

b) Provide the results of RICT sensitivity studies for AC and DC power-related TS LCOs submitted in the LAR that demonstrate the impact of not implementing the 24-hour PRA mission time. Include a discussion of the impact of the split mission times for the EDGs on the RICT calculations.

APLA QUESTION 02 – Probabilistic Risk Assessment Modeling of RCIC Black Start

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that Table 2-1 of PRA Notebook PB-MISC-043 states, "Systems that normally require DC [power] for operation are not credited for continued operation upon battery depletion"; however, the reactor core isolation cooling system (RCIC) is credited after battery depletion, referred to as "RCIC black start." The analysis in Table 2-1 states that the initial operation of RCIC or high-pressure coolant injection (HPCI) for 2 hours will provide sufficient reactor pressure vessel level to perform the RCIC black start prior to core damage. The analysis assessment states the RCIC black start credit represents "a slight conservative bias." It is unclear to the NRC staff whether this action is feasible, since the operators have no indication of vessel level or injection flow, and this is a conservative assumption. Provide the following information:

a) Identify which RICT TS LCOs are affected by the credit for RCIC black start.

b) Provide the basis for the feasibility of crediting RCIC after battery depletion (i.e., RCIC black start). Include in this discussion what licensee program directs this action (e.g., emergency operating procedures, severe accident management guidelines, mitigating FLEX strategies).

c) Provide the results of RICT sensitivity studies of the associated TS LCOs identified in part (a) that remove credit for RCIC black start. Include a discussion of the impact of this assumption on the RICT calculations.

APLA QUESTION 03 – Probabilistic Risk Assessment Modeling of High-Pressure Service Water

Table E1-1 of LAR Enclosure 1 regarding TS LCO 3.7.1.A (one high-pressure service water (HPSW), subsystem inoperable) states in Note 4 of the table that the HPSW consists of two independent subsystems. Each subsystem contains two HPSW pumps that discharge to both residual heat removal (RHR) heat exchangers. The design-success criteria (DSC) for this TS LCO in Table E1-1 is one of two subsystems; however, the PRA success criteria is one pump and one heat exchanger. It is unclear to the NRC staff whether the PRA success criteria is equivalent to a single subsystem as described in Note 4.

Provide a description of the HPSW system modeling in the Peach Bottom PRAs, and describe the analysis performed to support the PRA success criteria for HPSW.

APLA QUESTION 04 – Probabilistic Risk Assessment Modeling of Vacuum Breakers (Implementation Items)

LAR Attachment 6 lists the following implementation items that must be completed prior to implementation of the RICT program to satisfy the guidance in NEI 06-09 that the PRA reflect the as-built, as-operated plant and that the PRA technical adequacy is acceptable:

- Exelon will ensure that the reactor building-to-suppression chamber vacuum breakers are modeled in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.

- Exelon will ensure that the suppression chamber-to-drywell vacuum breakers are modeled in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.

LAR Attachment 6 also states that if implementation of any of these changes constitutes a PRA upgrade as defined in the PRA standard, as endorsed by RG 1.200, then a focused-scope peer review will be performed on these changes, and any findings will be resolved and incorporated in the PRA prior to the implementation of the RICT program. However, it is unclear to the NRC staff how the addition of these system models will meet CC-II of the PRA standard, as endorsed by RG 1.200. In light of these observations, provide the following information:

Regarding the implementation items identified above, describe how the associated systems will be adequately modeled in the PRA to CC-II. Include in this discussion:

i. How mechanical components, instrument channels, logic components, and other relevant system components will be modeled.

ii. Provide details of the success criteria for these systems. If the PRA success criteria do not match the DSC, then provide a justification for the PRA success criteria.

iii. Confirm whether these implementation items apply to both the internal events PRA (IEPRA) and the fire PRA (FPRA). Accordingly, adjust the wording for each of the affected implementation items in LAR Attachment 6. If any of these implementation items will not be applied to the FPRA, then justify the position that the FPRA model will be sufficient to support the RICT program.

APLA QUESTION 05 – Probabilistic Risk Assessment Modeling and Uncertainty of FLEX Equipment and 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 challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decisionmaking in accordance with the guidance of RG 1.200.

Regarding equipment failure probability in the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 8):

> The uncertainty associated with failure rates of portable equipment should be considered in the PRA models consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200. Risk-informed applications should address whether and how these uncertainties are evaluated.

Regarding human reliability analysis (HRA), NEI 16-06, Section 7.5, recognizes that the current HRA methods do not translate directly to human actions required for implementing mitigating strategies. Sections 7.5.4 and 7.5.5 of NEI 16-06 describe such actions to which the current HRA methods cannot be directly applied, such as debris removal, transportation of portable equipment, installation of equipment at a staging location, routing of cables and hoses, and those complex actions that require many steps over an extended period, multiple personnel and locations, evolving command and control, and extended time delays. In the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 11):

> Until gaps in the human reliability analysis methodologies are addressed by improved industry guidance, [human error probabilities] HEPs associated with actions for which the existing approaches are not explicitly applicable, such as actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06, along with assumptions and assessments, should be submitted to NRC for review.

Regarding uncertainty, Section 2.3.4 of NEI 06-09 states that PRA modeling uncertainties shall be considered in the application of the PRA base model results to the RICT program and that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation. NEI 06-09 also states that the insights from the sensitivity studies should be used to develop appropriate risk management actions (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 PRA modeling of FLEX related to the equipment 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 the RICTs proposed in this application.

LAR Enclosure 9, Table E9-1, indicates that FLEX equipment and actions have been credited in the IEPRA. The LAR states that a sensitivity study was performed for the IEPRA to address this issue. The LAR stated that the sensitivity did not significantly impact the RICT values. As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that Section 8 of PRA Notebook PB-MISC-043 provided results of a sensitivity study where the failure probability of the FLEX injection pump and diesel generator was significantly increased. However, the NRC staff notes the significant challenges of modeling FLEX equipment and actions without sufficient industry data and without a consensus HRA approach to address unique aspects of FLEX actions.

The NRC staff also notes that the difference between failure rates associated with permanently installed safety-related diesel generators and portable non-safety-related diesel generators could be greater than a factor of 10 without consideration of further uncertainty. It is unclear to the NRC staff whether the stated sensitivity study addressed the uncertainties associated with estimating HEP values for FLEX actions, especially for non-operator trained actions. Given the observations above, it is not clear whether the sensitivity study performed to assess the impact of crediting FLEX equipment and actions is sufficient to conclude that the impact to the RICT program of the uncertainties associated with modeling FLEX is negligible. For this reason, and to understand the credit that will be taken for FLEX equipment and actions in the RICT program, address the following separately for the IEPRA, internal flooding PRA, and FPRA:

a) Provide results of LCO-specific sensitivity studies that assess the removal of FLEX credit on RICT calculations.

b) Regarding HRA, address the following items:

   i. Discuss whether any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and Sections 7.5.5 of NEI 16-06.

      If any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and Sections 7.5.5 of NEI 16-06, answer either item (ii) or (iii) below.

   ii. Justify and provide results of LCO-specific sensitivity studies that assess impact from the FLEX-independent and FLEX-dependent HEPs associated with deploying and staging FLEX portable equipment on the RICTs proposed in this application. As part of the response, include the following information:

      1. Justify independent and joint HEP values selected for the sensitivity studies, including justification of why the chosen values constitute bounding realistic estimates.

      2. Provide numerical results on specific selected RICTs and discussion of the results.

3. Discuss composite sensitivity studies of the RICT results to the operator action HEPs and the FLEX equipment reliability uncertainty sensitivity study.

4. Describe how the source of uncertainty due to the uncertainty in FLEX operator action HEPs will be addressed in the RICT program. Describe specific RMAs being proposed and how these RMAs are expected to reduce the risk associated with this source of uncertainty.

iii. Alternatively to item (b)ii above, provide information associated with the following items listed in supporting requirements (SR) HR-G3 and HR-G7 of the PRA standard to support the NRC staff's detailed review of the LAR:

1. the level and frequency of training that the operators and non-operators receive for deployment of the FLEX equipment (performance shaping factor (a) in SR HR-G3),

2. performance shaping factor (f) in SR HR-G3 regarding estimates of time available and time required to execute the response,

3. performance shaping factor (g) in SR HR-G3 regarding complexity of detection, diagnosis, and decisionmaking and executing the required response,

4. performance shaping factor (h) in SR HR-G3 regarding consideration of environmental conditions, and

5. human action dependencies as listed in SR HR-G7 of the PRA standard.

c) The 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 the 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.

i. Provide an evaluation of the model changes associated with incorporating FLEX mitigating strategies that demonstrates that none of the following criteria are 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.

ii. Alternatively to item (c)i above, confirm that the modeling of FLEX equipment and FLEX actions in the PRA has been peer reviewed in accordance with NRC-accepted methods. Provide the findings of the peer review performed on the FLEX modeling and the disposition of the findings as they pertain to the impact on this LAR.

APLA QUESTION 06 – Probabilistic Risk Assessment Modeling and Uncertainty of Digital Instrumentation and Controls

Section 2.3.4 of NEI 06-09 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 states that this consideration is consistent with Section 2.3.5 of RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (ADAMS Accession No. ML100910008). NEI 06-09 further states that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties that could potentially impact the results of an RICT calculation and that sensitivity studies should be used to develop appropriate compensatory RMAs.

a) A TS LCO condition listed in LAR Table E1-1 indicates that instrumentation and control (I&C) modeling in the PRA is insufficient to model the condition, and therefore, the inoperability of the associated equipment (e.g., channel) will be modeled using a surrogate event. Furthermore, based on documentation in the LAR for other TS LCO conditions in the RICT program, it is not clear to NRC staff whether I&C is modeled in sufficient detail to support implementation of TSTF-505, Revision 2.

   Describe how I&C equipment that is applicable or that impacts the RICT calculations is modeled/considered in the PRA. Include in this discussion: (1) the scope of the I&C equipment that is explicitly modeled (e.g., bistables, relays, sensors, integrated circuit cards), (2) description of the level of detail that the PRA model supports (e.g., are all channels of an actuation circuit modeled), (3) discussion of the generic data and plant-specific data used, and (4) discussion of the associated TS functions for which an RICT can be applied.

b) Regarding digital I&C, the NRC staff notes the lack of consensus industry guidance for modeling these systems in plant PRAs to be used to support risk-informed regulatory applications. In addition, known modeling challenges exist such as lack of industry data for digital I&C components, differences between digital and analog system failure modes, and the complexities associated with modeling software failures, including common cause software failures. Also, although reliability data from vendor tests may be available, this source of data is not a substitute for in-the-field operational data. Given these challenges, the uncertainty associated with modeling a digital I&C system could impact the RICT program.

   Attachment 4 of the LAR identifies digital feedwater control system is employed at the plant. However, the modeling of this digital system is not identified in Enclosure 9 as a source of uncertainty. Therefore, it is not clear to the NRC staff whether the digital feedwater control system is the only digital system credited in the PRA and whether there are other digital systems credited in the PRA that could potentially impact RICT calculations. In light of these observations, provide the following information:

   i. Describe and provide the results of a sensitivity study performed for each digital system modeled in the PRA demonstrating that the uncertainty associated with PRA modeling the digital I&C systems has inconsequential impact on the RICT calculations.

   ii. As an alternative to item (b)i above, identify which LCOs are determined to be impacted by digital I&C system modeling for which RMAs will be applied during an

RICT. Explain and justify the criteria used to determine what level of impact to the RICT calculation requires additional RMAs.

APLA QUESTION 07 – PRA Update Process

Section 2.3.4 of NEI 06-09 specifies, "criteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations."

LAR Enclosure 7 states that if a plant change or a discovered condition is identified and can have significant impact on the RICT calculations, then an unscheduled update of the PRA models will be implemented. More specifically, the LAR states that if the plant changes meet specific criteria defined in the plant PRA and update procedures, including criteria associated with consideration of the cumulative risk impact, then the change will be incorporated into applicable PRA models without waiting for the next periodic PRA update. The LAR does not explain under what conditions an unscheduled update of the PRA model will be performed or the criteria defined in the plant procedures that will be used to initiate the update.

In light of these observations, describe the conditions under which an unscheduled PRA update (i.e., more than once every two refueling cycles) would be performed and the criteria that would be used to require a PRA update. In the response, define what is meant by "significant impact to the RICT Program calculations."

APLA QUESTION 08 – Real-Time Risk Model

Regulatory Position 2.3.3 of RG 1.174 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 describes attributes of the tool used for configuration risk management (CRM). Some 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 that vary by time of year or time in fuel cycle or otherwise demonstrate to be conservative or bounding.

- Application-specific risk important uncertainties contained in the CRM model (that are identified via PRA model to CRM tool 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 an 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.

Enclosure 8 of the LAR describes the attributes of the real-time-risk (RTR) model (i.e., Peach Bottom's CRM tool) for use in RICT calculations. The LAR explains that the internal events, internal flooding events, and fire events PRA models are maintained as separate models. The LAR also describes several changes made to the PRA models to support calculation of configuration-specific risk and mentions approaches for ensuring the fidelity of the RTR to the PRAs, including RTR maintenance, documentation of changes, and testing. Regarding development and application of the RTR model, provide the following information:

a) Describe the process that will be used to maintain the accuracy of any presolved cutsets with changes in plant configuration.

b) Describe the benchmarking activities performed to confirm consistency of the RTR model results to the results of each PRA model of record, including periodicity of RTR updates compared to the model of record updates. Address each model of record (i.e., internal events, internal flooding events, and internal fire events) in the response.

APLA QUESTION 09 – Impact of Seasonal Variations on the Real-Time Risk Model

Regulatory Position 2.3.3 of RG 1.174 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. Additionally, NEI 06-09 states the following:

> If the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle (examples include moderator temperature coefficient, summer versus winter alignments for HVAC, seasonal alignments for service water), then the RICT calculation shall either 1) use the more conservative assumption at all time, or 2) be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA.

Section 2 of LAR Enclosure 8 states, "The impact of outside temperatures on system requirements are addressed in the RTR model." As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff noted that PRA Notebook PB-MISC-043 states that two EDG fans are required when ambient temperature is above 80 degrees Fahrenheit (°F); however, the PRA model uses a split fraction to represent the percentage of the year assumed to be over 80 °F in modifying the success criteria. The analysis states that RICT will necessitate identifying specific time periods when two fans are required.

Provide further explanation supporting the statement above by summarizing the plant equipment subject to seasonal variations and how it is modeled in the PRA to remove the seasonal dependency.

APLA QUESTION 10 – Probabilistic Risk Assessment Model Uncertainty Analysis Process

The NRC staff SE to NEI 06-09 specifies that the LAR should identify key assumptions and sources of uncertainty and assess and disposition each as to its impact on the RMTS application. Section 5.3 of NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking, Main Report," dated March 2017 (ADAMS Accession No. ML17062A466), presents guidance on the process of identifying, characterizing, and qualitatively screening model uncertainties.

LAR Enclosure 9 states that the process for identifying key assumptions and sources of uncertainty for the IEPRA (includes internal floods) and FPRA was performed using the guidance in NUREG-1855, Revision 1. The LAR indicates that in addition to reviewing generic industry sources of uncertainty for applicability, the IEPRA and FPRA models and notebooks were reviewed for plant-specific assumptions and sources of uncertainty.

However, for the IEPRA (includes internal floods), it is not clear to the NRC staff what specific process and criteria were used to screen uncertainties from an initial comprehensive list of assumptions and sources of PRA modeling uncertainty (including those associated with plant-specific features, modeling choices, and generic industry concerns) in order to conclude that no uncertainty issues could impact the RICT calculations. The NRC staff notes from review of Enclosure 9 of the LAR that the dispositions to many identified sources of uncertainty highlight the phrase "not significantly impact the RICT values." It is not clear to the NRC staff what this phrase means in all cases. Also, for certain sources of uncertainty, the disposition states that a sensitivity study was performed to evaluate the impact of the uncertainty, but it is not clear what criteria was used to determine when a sensitivity study was performed or when additional RMAs should be considered.

Therefore, address the following regarding the IEPRA (includes internal floods) uncertainties:

a) Describe the process used to screen uncertainties from the initial comprehensive lists of PRA uncertainties (including those associated with plant-specific features, modeling choices, and generic industry concerns) in order to eventually conclude that the uncertainty issues could not impact the RICT calculations. Include a description of the criteria that was used to screen down from a comprehensive listing of sources of uncertainty to a smaller set of key candidate assumptions and sources of uncertainty. Also, describe the criteria used to justify that none of the key candidate assumptions and sources of uncertainty could have an impact on the RICT calculations. As part of this description, explain the criteria used to determine when the results of sensitivity studies do not significantly impact RICT values.

b) Concerning the evaluation criteria used to evaluate and screen uncertainties addressed in item (a) above:

    i. Discuss the criteria used to consistently determine when a sensitivity study was used to address the identified source of uncertainty.

    ii. Discuss the criteria used to consistently determine when additional RMAs should be implemented because of modeling uncertainty.

APLA QUESTION 11 – Performance Monitoring and Feedback

Section 2.3 of LAR Attachment 1 states that the application of an RICT will be evaluated using the guidance provided in NEI 06-09, which was approved by the NRC on May 17, 2007 (ADAMS Accession No. ML071200238). The NRC SE for NEI 06-09, states, "The impact of the proposed change should be monitored using performance measurement strategies." NEI 06-09 considers the use of NUMARC 93-01, Revision 4F, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (ADAMS Accession No. ML18120A069), as endorsed by RG 1.160, Revision 4 (ADAMS Accession No. ML18220B281), for the implementation of the Maintenance Rule. NUMARC 93-01, Section 9.0, contains guidance for the establishment of performance criteria.

Furthermore, Section 2.3 of LAR Attachment 1 states:

> In addition, the NEI 06-09-A, Revision 0 methodology satisfies the five key safety principles specified in Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications," dated August 1998 (ADAMS Accession No. ML003740176), relative to the risk impact due to the application of a RICT.

NRC staff position C.3.2 provided in RG 1.177 for meeting the fifth key safety principle acknowledges the use of performance criteria to assess degradation of operational safety over a period of time. It is unclear to the NRC staff how the licensee's process for the risk-informed application captures performance monitoring for the structures, systems, and components (SSCs) within-scope of the application. In light of these observations, address either (a) or (b) below.

a) Confirm that the Peach Bottom Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in the NRC-endorsed guidance in NUMARC 93-01.

   OR

b) Describe the approach/method used by Peach Bottom for SSC performance monitoring as described in Regulatory Position C.3.2 referenced in RG 1.177 for meeting the fifth key safety principle. In the description, include criteria (e.g., qualitative or quantitative), along with the appropriate risk metrics, and explain how the approach and criteria demonstrate the intent to monitor the potential degradation of SSCs in accordance with the NRC SE for NEI 06-09.

Probabilistic Risk Assessment Licensing Branch B (APLB) Audit Questions

RG 1.200 states that "NRC reviewers... [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application." Relatively extensive and detailed reviews of FPRAs were undertaken in support of each LAR to transition to National Fire Protection Association Standard 805 (NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants"). These reviews determined that implementation of some of the complex FPRA methods often used nonconservative and oversimplified assumptions to apply the method to specific plant configurations. Some of these issues were not always identified in F&Os by the peer review teams, but are considered potential key assumptions by the NRC staff. Using more defensible and less simplified assumptions could substantively affect the fire risk and fire risk profile of the plant.

The NRC staff evaluates the acceptability of the PRA for each new risk-informed application and, as discussed in RG 1.174, recognizes that the technical acceptability of risk analyses necessary to support regulatory decisionmaking may vary with the relative weight given to the risk assessment element of the decision-making process. The NRC staff notes that the calculated results of the PRA are used to calculate an RICT, which subsequently determines how long SSCs (both individual SSCs and multiple, unrelated SSCs) controlled by TSs can remain inoperable. Therefore, the PRA results are given a very high weight in a TSTF-505 application, and the NRC staff requests additional information on the following issues that have been previously identified as potentially key FPRA assumptions.

APLB QUESTION 01 – Reduced Transient Heat Release Rates

The key factors used to justify using transient fire-reduced heat release rates (HRRs) below those prescribed in NUREG/CR-6850, "EPRI/NRC Fire PRA Methodology for Nuclear Power Facilities" (ADAMS Accession No. ML052580075), are discussed in a letter from the NRC to NEI, dated June 21, 2012 (ADAMS Package Accession No. ML12172A406).

If any reduced transient HRRs below the bounding 98th percentile HRR of 317 kilowatts (kW) from NUREG/CR-6850 were used, discuss the key factors used to justify the reduced HRRs. In this discussion, also provide the following information:

a) Identification of the fire areas where a reduced transient fire HRR is credited and what reduced HRR value was applied.

b) A description for each location where a reduced HRR is credited, and a description of the administrative controls that justify the reduced HRR, including how location-specific attributes and considerations are addressed. Include a discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance. Also, include discussion of the personnel traffic that would be expected through each location.

c) The results of a review of records related to compliance with the transient combustible and hot work controls.

APLB QUESTION 02 – Joint Human Error Probability Floor

NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (ADAMS Accession No. ML12216A104), discusses the need to consider a minimum value for the joint probability of multiple human failure events (HFEs) in HRAs. NUREG-1921 refers to Table 2-1 of NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)" (ADAMS Accession No. ML051160213), which recommends that joint human error probability (HEP) values should not be below 1E-5. Table 4-4 of Electric Power Research Institute (EPRI) 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. Therefore, the guidance in NUREG-1921 allows for assigning joint HEPs that are less than 1E-5, but only through assigning proper levels of dependency.
TSTF-505 evaluations use the FPRA and IEPRA. The LAR does not provide information about whether and, if so, what minimum joint HEP value(s) is currently assumed in the FPRA. Also,

even if the assumed minimum joint HEP value(s) is shown to have no impact on the current FPRA risk estimates, it is not clear to the NRC staff how it will be ensured that the impact remains minimal for future PRA revisions. In light of these observations, provide the following information:

a) Explain what minimum joint HEP value(s) was assumed in the FPRA.

b) If a minimum joint HEP value less than 1E-05 was used in the FPRA, then provide a description of the sensitivity study that was performed and the quantitative results that justify that the minimum joint HEP value(s) has no impact on the RICT application.

c) If, in response to item (b) above, it cannot be justified that the minimum joint HEP value(s) has no impact on the application, confirm that each joint HEP value used in the FPRA below 1E-5 includes its own separate justification that demonstrates the inapplicability of the NUREG-1792 lower value guideline (i.e., using such criteria as the dependency factors identified in NUREG-1921 to assess level of dependence). Provide an estimate of the number of these joint HEP values below the guideline value of 1E-5 for the FPRA, discuss the range of values, and provide at least two different examples where this justification is applied.

APLB QUESTION 03 – Obstructed Plume Model

NUREG-2178, Volume 1, “Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE)” (ADAMS Accession No. ML16110A140), contains refined peak HRRs, compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction. Additionally, NUREG-2178 provides guidance that indicates that the obstructed plume model is not applicable to cabinets in which the fire is assumed to be located at elevations of less than one-half of the cabinet height.

a) If obstructed plume modeling was used, then indicate whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet height.

b) Justify any modeling in which the base of an obstructed plume is located at less than one-half of the cabinet’s height.

APLB QUESTION 04 – Systems Not Credited in the Fire PRA

As part of its audit (ADAMS Accession No. ML20217L346), the NRC staff reviewed PRA Notebook PB-PRA-021.62, which noted that several systems were identified as not being modeled in the FPRA. The NRC staff notes that some conservative PRA modeling assumptions could have a nonconservative impact on the RICT calculations. If an SSC is part of a system not credited in the FPRA or is supported by a system that is assumed to always fail, then the risk increases due to taking that SSC out of service are masked. Therefore, provide the following information:

a) Identify the systems or components that are assumed to be always failed in the FPRA or not included in the FPRA (e.g., due to lack of cable tracing or other reasons). Justify that these assumptions have an inconsequential impact on the RICT calculations and no RMAs are required to address these items.

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 that accounts for the impact on the RICT of the 1) conservative FPRA assumption of failed SSCs or 2) SSCs not included in the FPRA model. The proposed mechanism should also ensure that any additional risk from correcting these assumptions is either accounted for in the RICT calculations or is compensated for by applying additional RMAs during the RICT.

APLB QUESTION 05 - Implementation Item for Cable Data for Standby Liquid Control

LAR Attachment 6 lists the following implementation item that must be completed prior to implementation of the RICT program to satisfy the guidance in NEI 06-09 that the PRA reflect the as-built, as-operated plant and that the PRA technical adequacy is acceptable:

- Exelon will ensure that the updated standby liquid control cable data will be incorporated in the Peach Bottom PRA with sufficient detail to accurately calculate the RICT.

LAR Attachment 6 also states that if implementation of this change constitutes a PRA upgrade as defined in the PRA standard, as endorsed by RG 1.200, then a focused-scope peer review will be performed on this change, and any findings will be resolved and incorporated in the PRA prior to the implementation of the RICT program. However, it is unclear to the NRC staff how the addition of this system model will meet CC-II of the PRA standard, as endorsed by RG 1.200.

In light of this observation, describe how this system will be adequately modeled in the FPRA and in accordance with the PRA standard's CC-II.

APLB QUESTION 06 – Well-Sealed Motor Control Center Cabinets

Guidance in Frequently Asked Question (FAQ) 08-0042, "Fire Propagation from Electrical Cabinets" (from Supplement 1 of NUREG/CR-6850), applies to electrical cabinets below 440 volts (V). With respect to Bin 15 as discussed in Chapter 6 of NUREG/CR-6850, it clarifies the meaning of "robustly or well-sealed." Thus, for cabinets of 440 V or less, fires from well-sealed cabinets do not propagate outside the cabinet. For cabinets of 440 V and higher, the original guidance in Chapter 6 remains and requires that Bin 15 panels, which house circuit voltages of 440 V or greater, are counted, because an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires).

FAQ 14-0009, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V" (ADAMS Accession No. ML15119A176), provides the technique for evaluating fire damage from MCC cabinets having a voltage greater than or equal to 440 V. Therefore, propagation of fire outside the ignition source panel must be evaluated for all MCC cabinets that house circuits of 440 V or greater.

a) Describe how fire propagation outside of well-sealed MCC cabinets greater than or equal to 440 V is evaluated.

b) If well-sealed cabinets less than 440 V are included in the Bin 15 count of ignition sources, provide justification for using this approach, as this is contrary to the guidance.

APLB QUESTION 07 – Fire Probabilistic Risk Assessment Method for Very Early Warning Fire Detection Systems

LAR Enclosure 9, Section 4, states that the Peach Bottom FPRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by the NRC. Part (e) of TS 5.5.16 states that the approaches and methods used in the RICT program shall be acceptable to the NRC. Methods to assess risk must be those used to support the LAR or other methods approved by the NRC for generic use.

There have been some changes to the FPRA methodology since the development of the Peach Bottom FPRA that was peer reviewed. The integration of the NRC-accepted FPRA method described in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE)" (ADAMS Accession No. ML16343A058), may be relevant to this submittal and could potentially impact the TSTF-505 results, CDF, or large early release frequency (LERF).

Section 2.5.5 of RG 1.174 provides guidance that indicates additional analysis is necessary to ensure that contributions from the above influence would not change the conclusions of the LAR.

a) If the above guidance has not been implemented in the Peach Bottom FPRA, provide a detailed justification for why the integration of this guidance would not change the conclusions of the LAR and subsequently not impact the TSTF-505 RICT calculations and risk metrics for total CDF and total LERF. As part of this justification, identify any FPRA methodologies used in the FPRA that are no longer accepted by the NRC staff (e.g., guidance provided in FAQ 08-0046, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems" (ADAMS Accession No. ML093220426), has been retired by letter dated July 1, 2016 (ADAMS Accession No. ML16167A444)). Provide technical justification for its use in TSTF-505 RICT calculations and evaluate the significance of its use on the RICT estimates.

   OR

b) Alternatively, if the above guidance has been implemented in the FPRA, provide the following information:

   i. Indicate whether the changes to the FPRA are PRA maintenance or a PRA upgrade as defined in the PRA standard, Section 1-5.4, as qualified by RG 1.200, along with justification for this determination.

   ii. Discuss any focused- or full-scope peer reviews performed to evaluate these changes that were determined in item (b)(i) above to constitute a PRA upgrade, including when the peer review was performed and when the peer review report that evaluated the upgrade was approved.

APLB QUESTION 08 – Treatment of Sensitive Electronics

FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (ADAMS Accession No. ML13322A085), provides supplemental guidance for application of the damage criteria provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850, Volume 2, for solid-state and sensitive electronics.

a) Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in Frequently Asked Question (FAQ) 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).

b) If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the RICT calculations.

APLB QUESTION 09 – Probabilistic Risk Assessment Treatment of Dependencies Between Units 2 and 3

Many plants have multiple units adjoined and thus have common areas. For these plants, the risk contribution from fires originating in one unit must be addressed for impacts to the other unit, given the physical proximity of the other unit, common areas, and the existence of shared systems. Therefore, address the following if Units 2 and 3 have common areas and shared systems:

a) Explain how the risk contribution of fires originating in one unit is addressed for the other unit, given impacts due to the physical proximity of equipment and cables in one unit to equipment and cables in the other unit. Include identification of locations where a fire in one unit can affect components in the other unit, and explain how the risk contributions of such scenarios are allocated for an RICT calculation.

b) Explain how the contributions of fires in common areas are addressed, including the risk contribution of fires that can impact components in both units.

c) Explain the extent to which systems are shared by both units and whether shared systems are credited in the PRA models (IEPRA and FPRA) for both units. If shared systems are credited in the PRA models for each unit, then explain how the PRAs address the possibility that a shared system is demanded in both units in response to a single internal events initiating event or fire initiator.

APLB QUESTION 10 - Probabilistic Risk Assessment Model Uncertainty Analysis Results

The NRC staff SE to NEI 06-09 specifies that the LAR should identify key assumptions and sources of uncertainty and should assess/disposition each as to its impact on the RMTS application. LAR Enclosure 9, Table E9-3, identifies the key assumptions and sources of uncertainty for the FPRA and provides dispositions for each source of uncertainty for this TSTF-505 application. The NRC staff reviewed the dispositions provided in LAR Table E9-3 to the key assumptions and sources of modeling uncertainty and noted that not all uncertainties that appeared to have the potential to impact the RICT calculations seemed fully resolved.

LAR Enclosure 9, Table E9-3, identifies post-fire HRA as a source of FPRA modeling uncertainty because fire HEPs must be adjusted to consider the additional challenges present given a fire. The LAR states that industry consensus modeling approaches are used and concludes that this source of uncertainty impact "is expected to be small" with apparently no sensitivities being performed. To address this source of uncertainty, the LAR states that appropriate RMAs would be required – for example, pre-job briefs. It is unclear to the NRC staff how the RMAs will adequately address the impact on RICT values. Therefore, address the following items:

a) Justify that the uncertainty associated with post-fire HRA modeling does not have a consequential impact on calculated RICTs for components supporting TS LCO conditions in the RICT program.

OR

b) Explain what RMAs will be considered to compensate for this uncertainty.

APLB QUESTION 11 – Fire Modeling

The LAR referred to risk evaluation and the application of fire modeling technology. The NRC staff was unable to fully evaluate the fire modeling performed as part of the FPRA.

Regarding the acceptability of the FPRA approach, methods, and data, describe the fire modeling calculational model or numerical methods (e.g., fire modeling tools and techniques) used in support of the FPRA.

APLB QUESTION 12 – Damage Thresholds

Part 4 of ASME/ANS RA-Sa-2009 indicates that damage thresholds be established to support the FPRA. The PRA standard further indicates that thermal impact(s) must be considered in determining the potential for thermal damage of SSCs, and appropriate temperature and critical heat flux criteria must be used in the analysis. Therefore, provide the following information:

a) Describe how the installed cabling in the fire areas was characterized, specifically regarding the critical damage threshold temperatures and critical heat fluxes for thermoset and thermoplastic cables.

b) An IEEE-383 (Institute of Electrical and Electronics Engineers Standard 383, "IEEE Standard for Type Test of Class 1 E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations") qualified cable may or may not meet the criteria for a 'thermoset cable." It is also possible that a non-IEEE-383 qualified cable actually meets the criteria for a "thermoset" cable. Provide clarification on the assumptions that were made in terms of damage thresholds of cables.

c) For those areas that are assumed to have thermoset damage criteria, confirm that the cables are actually thermoset and that the potential confusion about IEEE-383/thermoset is not applicable.

d) Describe how raceways with a mixture of thermoplastic and thermoset cables are treated in terms of damage thresholds.

e) In each fire area where they are credited, explain how cable tray covers, fire-resistant coatings, and fire wraps were credited in terms of delaying or preventing damage of cables. In addition, explain how holes in cable tray covers were treated regarding the fire modeling damage criteria.

f) Explain how the damage thresholds for non-cable components (i.e., pumps, valves, electrical cabinets, etc.) were determined. Identify any non-cable components that were assigned damage thresholds different from those for thermoset and thermoplastic cables, and provide a technical justification for these damage thresholds.

Probabilistic Risk Assessment Licensing Branch C (APLC) Audit Questions

APLC QUESTION 01 – Impacts from Seismic Hazard Frequencies

Section 2.3.1, Item 7 of NEI 06-09, 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 SE for NEI 06-09 states that "Where PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT."

In Section 3 of Enclosure 4 to the LAR, the licensee stated that the site-specific seismic PRA (SPRA) completed in response to the 10 CFR 50.54(f) request for information associated with the Fukushima Near-Term Task Force (NTTF) activities is not directly used in the RICT program but provides input into the calculation for seismic core damage frequency (SCDF) and seismic large early release frequency (SLERF). The licensee selected the seismic hazard curve that was used in the development of NTTF SPRA model, which is based on the peak ground acceleration (PGA). In the same section of the LAR, the licensee mentioned its seismic hazard and screening report (ADAMS Accession No. ML14090A247), which provided seismic hazard curves at various frequencies at 100 (PGA), 25, 10, 5, 2.5, 1, and 0.5 hertz (Hz). The NRC staff compared the seismic hazard curves between these two documents and found that the PGA hazard curve used in the LAR is different than that in the seismic hazard and screening report.

a) Explain the difference between the two PGA hazard curves cited above and justify the selection of the PGA hazard curve for use in the estimation of the SCDF penalty in the LAR.

b) Justify that the consideration of seismic hazard curves at frequencies other than the PGA does not significantly change the SCDF penalty proposed in the LAR.

APLC QUESTION 02 – Representativeness of Discretization of Seismic Hazard Curve

The licensee provided the PGA seismic hazard curve data from 0.005 gram (g) to 7.5 g in Table E4-1 of Enclosure 4 to the LAR. The seismic hazard interval frequencies are represented by discretizing the hazard curve into eight 'bins' as shown in Table E4-2 of Enclosure 4 to the LAR. The representative PGA for the last 'bin' is selected to be 0.99 g for representing the entire hazard from 0.9 g to 7.5 g. This approach results in a mean fragility probability of 0.95 instead of 1.0 as shown in Table E4-3 of Enclosure 4 to the LAR. As explained in Enclosure 4 to the LAR, this change has a minor impact on the estimated SCDF value. However, the NRC staff notes that sensitivity analysis 1d in the licensee's SPRA report (ADAMS Accession No. ML18240A065) shows a 17 percent increase in SLERF due to refinement in the discretization of the last 'bin.' This is likely to increase the seismic conditional large early release probability (SCLERP) estimate, and therefore, the SLERF penalty estimate. The LAR does not discuss the impact of the refinement of the discretization for the last 'bin' on the estimated SLERF penalty.

Justify that the selected representative PGA of 0.99 g for the last ‘bin’ is reasonable and conservative for the estimated SLERF penalty or provide an updated SLERF penalty.

APLC QUESTION 03 – Seismic Core Damage Frequency and Large Early Release Frequency Penalty Estimate

Section 2.3.1, Item 7 of NEI 06-09, 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 SE for NEI 06-09 states that “Where PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT.”

The seismic penalty approach is used to quantify the risk impact and to support the RICT evaluation. The staff notes that there is a site-specific seismic PRA that could be used for this analysis. Section 3 of Enclosure 4 to the LAR states that the site-specific SPRA was not directly used in the RICT program but provided input into the calculation for SCDF and SLERF. The licensee compared the estimated SCDF penalty for the proposed RICT calculations against the point-estimate SCDF from the site-specific SPRA. In addition, the licensee used the SLERF to SCDF ratio from the site-specific SPRA to determine the SLERF penalty for use in the proposed RICT calculations.

The comparison of the estimated SCDF and SLERF penalties against the corresponding point-estimate mean values from the site-specific SPRA does not provide justification that the SCDF and SLERF penalty estimates are conservative, as stated in the NEI 06-09 guidance. There is no upper bound on the change-in-risk calculation, and the change in risk can exceed the base SCDF and SLERF. However, it appears to the NRC staff that the SPRA could provide the means to justify that the proposed SCDF and SLERF penalty estimates are conservative, and therefore, consistent with the staff’s SE for NEI 06-09.

Justify that the SCDF and SLERF penalty estimates are conservative based on the results and insights from change-in-risk calculations for the proposed RICTs using the recent site-specific SPRA.

<u>Technical Specifications Branch (STSB) Audit Questions</u>

STSB QUESTION 01 – Technical Specification 3.5.1.E, One ADS [Automatic Depressurization System] Valve Inoperable

LAR Enclosure 1, Table E1-1 lists in the column of “TS 3.5.1.E” a condition with one ADS valve inoperable. The corresponding column of the “SSCs Covered by TS LCO Condition” indicates that ADS (five safety relief valves) are required to be operable, and the column of “Design Success Criteria” indicates that five ADS valves are available.

Clarify for TS 3.5.1.E condition with one of five required ADS valves inoperable, that the Design Success Criteria need 3 or 4 available ADS valves. Discuss the Analyses of Record (AOR) that demonstrated adequacy of 3 or 4 ADS valves for reactor pressure vessel rapid depressurization to mitigate the loss-of-coolant accident consequences and reference the NRC documents approving the AOR of the concern or address the acceptability of the AOR if it was not previously approved by the NRC.

STSB QUESTION 02 – Technical Specification 3.5.1.F, One Automatic Depressurization System valve inoperable and One Low Pressure Emergency Core Cooling System Subsystem Inoperable

LAR Enclosure 1, Table E1-1 lists in the column of “TS 3.5.1.F” a condition with one ADS valve inoperable and one low pressure Emergency Core Cooling System (ECCS) injection/spray subsystem inoperable. Clarify the same for 3.5.1.F. The corresponding column of the “SSCs Covered by TS LCO Condition” states, “See LCO Condition 3.5.1.A and 3.5.1.E,” which indicates that ADS (five safety relief valves) are required to be operable, and the column of “Design Success Criteria” indicates that five ADS valves are available.

Clarify for TS 3.5.1.F Condition with one of 5 required ADS valves inoperable, that the DSC need three or four available ADS valves. Discuss the AOR that demonstrated adequacy of three or four ADS valves for reactor pressure vessel rapid depressurization to mitigate the loss-of-coolant accident consequences and reference the NRC documents approving the AOR of the concern, or address the acceptability of the AOR if it was not previously approved by the NRC.

Electrical Engineering Branch (EEEB) Audit Questions

EEEB QUESTION 01 – Technical Specification 3.8.1.D, Two or More Offsite Alternating Current Power Circuits Inoperable

Peach Bottom’s DSC is derived from the current licensing basis of the plant, as documented in the Updated Safety Analysis Report, and should include a minimum set of required equipment that has the capacity and capability to safely shut down the reactor in case of an accident and maintain it in a safe condition. In Table E1-1 of Enclosure 1 of the LAR, the DSC for TS LCO 3.8.1.D (two or more offsite AC power circuits inoperable) is “one of two offsite AC power sources.” The NRC staff notes that if both offsite circuits are inoperable, one offsite AC power source as listed in the DSC is not available to provide the necessary power to safely shut down the reactor and maintain it in safe condition. Therefore, it is not clear how one offsite circuit can be the DSC for TS 3.8.1.D during the RICT program entry.

Explain this apparent discrepancy in Table E1-1 of Enclosure 1 of the LAR. Additionally, describe any effect the discrepancy may have on the PRA success criteria for TS LCO 3.8.1.D.

EEEB QUESTION 02 – Technical Specification 3.8.1.B, One Diesel Generator Inoperable

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.1 Condition B is “three of four diesel generators.” Explain the basis for this DSC. Include in the explanation, as necessary to clarify the basis, a description of the onsite AC power system’s design configuration, including each diesel generator’s capacity and loading.

EEEB QUESTION 03 – Technical Specification 3.8.4 Conditions A, B, C, D, and E

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.4 Conditions A, B, C, D, and E is “three of four DC divisions.” Explain the basis for the DSC for these TS conditions. Include in the explanation for each applicable TS condition, as necessary to clarify the basis, a description of each unit’s 125 VDC and 250 VDC system’s configuration, including number and type of batteries and chargers with associated capacities and loading, and use of any cross ties, as applicable.

EEEB QUESTION 04 – Technical Specification 3.8.7 Conditions A, B, C, and D

Table E1-1 in Enclosure 1 of the LAR states that the DSC for TS 3.8.7 Conditions A, B, C, and D is “three of four divisions.” Explain the basis for the DSC for these TS conditions. Include in the explanation for each TS condition, as necessary to clarify the basis, a description of the associated system configuration.

EEEB QUESTION 05 – RMA Examples

As part of its evaluation, the NRC staff reviews the proposed RMA examples for reasonable assurance that the RMAs are considered to monitor and control risk and to ensure adequate defense in depth. Enclosure 12 of the LAR describes the RMAs examples for TS 3.8.1.A, TS 3.8.1.B, TS 3.8.1.D, and TS 3.8.4.A. However, the LAR does not include the RMA examples for TS 3.8.7 conditions related to the power distribution system. Provide the RMA example(s) for TS 3.8.7.

Instrumentation & Controls Branch B (EICB) Audit Questions

EICB QUESTION 01 – Instrumentation & Controls Redundancy and Diversity

RG 1.174, Revision 3, states the licensee should assess whether the proposed licensing basis change meets the defense-in-depth principle by not overrelying on programmatic activities as compensatory measures associated with the change in the licensing basis. RG 1.174 further elaborates that human actions (e.g., manual system actuation) are considered as one type of compensatory measure.

Therefore, in LAR Attachment 5, if the diverse means identified are the manual actuations, demonstrate by one example that these “manual actuations” identified as the diverse means are modeled in the plant PRA defined in plant operation procedures to which operators are trained, and confirm the completion times associated with these actions are evaluated as adequate.

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 − REGULATORY VIRTUAL AUDIT PLAN REGARDING LICENSE AMENDMENT REQUEST TO ADOPT TSTF-505, REVISION 2 (EPID L-2020-LLA-0120) DATED OCTOBER 21, 2020

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