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10 CFR 50.90

August 10, 2018

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

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Response to Request for Additional Information and Supplemental Information - Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

References:

1. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants,'" dated August 30, 2017 (ML17243A014)

2. Letter from Richard B. Ennis (U.S. Nuclear Regulatory Commission) to Bryan C. Hanson, Exelon Generation Company, LLC – "Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Adoption of title 10 of the Code of Federal Regulations Section 50.69," dated October 10, 2017 (ML17272B016)

3. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Supplement to Application to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," dated October 24, 2017 (ML17297B521)

4. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Draft 50.69 Request for Additional Information (RAIs) - Peach Bottom," dated March 21, 2018

5. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information - Adopt 50.69 License Amendment," dated April 6, 2018 (ML18096B506)

6. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Response to Request for Additional Information - Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," dated May 7, 2018 (ML18128A009)

7. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Supplemental Information to Support Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," dated June 6, 2018 (ML18157A260)

8. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information 2nd Round - Adopt 50.69 License Amendment," (Draft) dated June 26, 2018

9. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information 2nd Round - Adopt 50.69 License Amendment," dated July 10, 2018 (ML18192A119)

10. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information 2nd Round - Adopt 50.69 License Amendment," dated July 17, 2018 (ML18200A274)

By letter dated August 30, 2017 (Reference 1), Exelon Generation Company, LLC (Exelon) requested an amendment to the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively. The proposed amendment would modify the licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." In a letter dated October 10, 2017 (Reference 2), the U.S. Nuclear Regulatory Commission (NRC) requested that Exelon provide supplemental information in support of the application. By letter dated October 24, 2017 (Reference 3), Exelon responded to the NRC's request for supplemental information.

The NRC reviewed the information provided in the Reference 1 and 2 submittals and indicated the need for additional information in order to complete their review and evaluation of the amendment request. In an electronic mail message dated March 21, 2018 (Reference 4), the NRC issued a draft Request for Additional Information (RAI). This draft RAI was the subject of further discussions during a teleconference on April 6, 2018, between Exelon and NRC representatives and additional clarification was provided. The NRC formally issued the RAI on April 6, 2018 (Reference 5), and Exelon responded to the RAI by letter dated May 7, 2018 (Reference 6). By letter dated June 6, 2018 (Reference 7), Exelon provided supplemental information in support of the proposed license amendment request to adopt 10 CFR 50.69.

Subsequently, in an electronic mail message dated June 26, 2018 (Reference 8), the NRC issued a second RAI (draft) that was the subject of discussions during a teleconference on July 10, 2018, between Exelon and NRC representatives. The NRC formally issued the RAI via electronic mail message dated July 10, 2018 (Reference 9), and requested a response by August 10, 2018. This RAI was supplemented via a subsequent electronic mail message dated July 17, 2018 (Reference 10), to provide additional clarifying information to one of the questions.

Attachment 1 to this letter provides a restatement of the RAI questions cited in References 9 and 10 followed by Exelon's responses.

Exelon is also providing supplemental information to modify the wording for the proposed license conditions previously described in the Reference 7 submittal. Attachment 2 provides updated proposed markups of the PBAPS, Units 2 and 3, Renewed Facility Operating Licenses which supersede in their entirety the markups provide in the Reference 7 letter.

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.

There are no new regulatory commitments in this response.

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

If you have any questions or require additional information, please contact Richard Gropp at 610-765-5557.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 10th day of August 2018.

Respectfully,

David T. Gudger

David T. Gudger
Manager, Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments:

1. Response to Request for Additional Information - Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

2. Updated Markups of Proposed Renewed Facility Operating License (RFOL) Pages

cc: w/ Attachments
Regional Administrator - NRC Region I
NRC Senior Resident Inspector - Peach Bottom Atomic Power Station
NRC Project Manager, NRR - Peach Bottom Atomic Power Station
Director, Bureau of Radiation Protection – Pennsylvania Department
of Environmental Protection
R. R. Janati, Pennsylvania Bureau of Radiation Protection
D. A. Tancabel, State of Maryland

ATTACHMENT 1

License Amendment Request

Peach Bottom Atomic Power Station, Units 2 and 3
Docket Nos. 50-277 and 50-278

Response to Request for Additional Information - Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

Attachment 1

Response to Request for Additional Information - Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

By letter dated August 30, 2017 (Reference 1), Exelon Generation Company, LLC (Exelon) requested an amendment to the Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively. The proposed amendment would modify the licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." In a letter dated October 10, 2017 (Reference 2), the U.S. Nuclear Regulatory Commission (NRC) requested that Exelon provide supplemental information in support of the application. By letter dated October 24, 2017 (Reference 3), Exelon responded to the NRC's request for supplemental information.

The NRC reviewed the information provided in the Reference 1 and 2 submittals and indicated the need for additional information in order to complete their review and evaluation of the amendment request. In an electronic mail message dated March 21, 2018 (Reference 4), the NRC issued a draft Request for Additional Information (RAI). This draft RAI was the subject of further discussions during a teleconference on April 6, 2018, between Exelon and NRC representatives and additional clarification was provided. The NRC formally issued the RAI on April 6, 2018 (Reference 5), and Exelon responded to the RAI by letter dated May 7, 2018 (Reference 6). By letter dated June 6, 2018 (Reference 7), Exelon provided supplemental information in support of the proposed license amendment request to adopt 10 CFR 50.69.

Subsequently, in an electronic mail message dated June 26, 2018 (Reference 8), the NRC issued a second RAI (draft) that was the subject of discussions during a teleconference on July 10, 2018, between Exelon and NRC representatives. The NRC formally issued the RAI via electronic mail message dated July 10, 2018 (Reference 9), and requested a response by August 10, 2018. This RAI was supplemented via a subsequent electronic mail message dated July 17, 2018 (Reference 10), to provide additional clarifying information to one of the questions.

Below is a restatement of the questions cited in References 9 and 10 followed by Exelon's responses.

RAI 03.c.01 – Follow-up to Licensee Response to NRC RAI 03.c regarding Fire Modeling

The disposition to partially resolved Fact and Observation (F&O) 2012-1-40 states that detailed two-point fire modeling was not been performed for all risk significant scenarios but this treatment did not have a significant impact on the application. In RAI 03.c, the NRC staff requested that the licensee justify its statement that not fully resolving the F&O has a minimal impact on the application or to provide a mechanism that ensures that the two-point fire modeling is applied to risk significant fire scenarios prior to implementing the 10 CFR 50.69 categorization process. The response to RAI 03.c states that this finding will be resolved prior to

implementing 10 CFR 50.69. The licensee stated that "risk significant fire PRA scenarios capable of being modeled using a two-point modeling approach will be updated using a two (or more) point fire modeling approach" prior to implementation of the 50.69 process.

i. *Explain what the phrase "capable of being modeled using a two-point modeling approach" means and what type of fire scenarios will not be modeled using this approach.*

Response:

The latest guidance for modeling fires originating in cable tray and junction boxes in Frequently Asked Questions (FAQs) 13-0005 and 13-0006 does not include a two-point modeling approach. These scenarios, if risk-significant, will not be modeled using a two-point approach as the methodology specifies a single modeling approach that includes failure of the ignition source. Other risk significant fire scenarios will be modeled using a two (or more) point fire modeling approach.

ii. *Justify how supporting requirement (SR) FSS-C1 will be met at Capability Category (CC) II after the update is performed.*

Response:

Capability Category CC II of ASME/ANS PRA Standard supporting requirement FSS-C1 requires use of a probabilistic representation of the range of fire intensities and durations for risk significant ignition sources. After the update, risk significant scenarios will be modeled using a two-point modeling approach for ignition sources other than cable tray and junction box fires (for which the modeling will be in accordance with FAQs 13-0005 and 13-0006). For scenarios involving ignition sources for which Heat Release Rate (HRR) distributions are applicable, two (or more) points on the HRR distribution curves will be used to analyze the scenarios. Scenarios that do not use HRR distributions will be analyzed with multiple severities as presented in NUREG/CR-6850 and related FAQs. For example, risk significant scenarios involving oil spills will postulate varying oil spill sizes. This approach satisfies the CC II requirement of FSS-C1.

iii. *If SR FSS-C1 will not be met at CC-II, justify why not using a two-point model for all risk significant scenarios has no impact on the application.*

Response:

FSS-C1 will be met at CC-II prior to performing 50.69 categorization.

RAI 05.b.01 – Follow-up to Licensee Response to NRC RAI 05.b regarding Uncoordinated Breakers in the Fire PRA

The disposition to open F&O 2012-5-1 states that if the licensee identifies uncoordinated circuits, then the uncoordinated circuits will be modeled in the fire PRA. In RAI 05.b.ii, the NRC staff requested description of how modeling of uncoordinated circuits will be performed. The response to RAI 05.b proposed an implementation item stating that the uncoordinated circuits will be modelled or additional analysis will be performed to show that the circuits are coordinated. Confirm that any additional analysis performed to determine that circuits are coordinated will be performed in accordance with guidance in NUREG/CR-6850 on breaker coordination studies.

Response:

Exelon confirms that, if additional analysis is performed to show that circuits are coordinated, the analysis will be performed in accordance with the guidance provided in NUREG/CR-6850 on breaker coordination studies.

RAI 08.d.01 – Follow-up to Licensee Response to NRC RAI 08.d regarding the RCIC and HPCI Turbine Failure Probabilities

In RAI 08.d.ii the NRC staff requested the licensee to provide justification for the nominal failure probability assigned to the failure of the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) turbines to run after passing liquid. The licensee's response to RAI 08.d states that failure of the HPCI and RCIC turbines to start were assigned a nominal value of 0.05 based on operator and system manager interviews, and based on engineering judgement. However, the licensee stated that the "HPCI and RCIC turbines are not specifically designed to continue running while passing liquid" and there was no industry failure data or valid testing under accident conditions cited in the RAI response. Provide the following:

i. A sensitivity study or other quantitative justification which demonstrates that the 10 CFR 50.69 categorization results are not sensitive to the assumed 0.05 failure probability. NOTE: The industry PRA standard ASME/ANS RA-Sa-2009 states that a reasonable alternative assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1 (ML17062A466) states that for operating equipment reliability for which no specific calculations exist, an accepted conservative model assumption is to assign a failure probability of 1 for the equipment in question. Therefore, consistent with the above guidance, this sensitivity study should set the specific parameter of concern to a turbine failure probability to 1, or at a minimum, increase it by a factor of 10.

Response:

As noted in the original RAI 08.d response, manufacturer testing and substantial industry operational experience exist to demonstrate that conditions involving significant water carryover do not lead to turbine failure. Based on EPRI maintenance guide information (Reference 11 - 2013 EPRI Report 3002001669), a recoverable turbine overspeed trip would be expected given water carryover into the High Pressure Coolant Injection (HPCI) or Reactor Core Isolation Cooling (RCIC) systems' steam lines, which is the scenario included in the Probabilistic Risk Assessment (PRA) model. Based on industry reports on the subject and industry experience as described in the original RAI response, catastrophic unrecoverable turbine failure given scenarios where there may be water in the steam lines, as modeled in the PRA, is unlikely. Additionally, the HPCI and RCIC restart procedures for Peach Bottom (References 15 and 16) include steps to reset the overspeed trip if needed and open the steam exhaust drain lines prior to restarting the pumps. The RCIC procedure includes a specific caution to allow for appropriate steam line drainage. The scenarios of interest for the unrecoverable failure of HPCI/RCIC are those where the RPV level reaches the main steam line. In such scenarios, there would be a significant amount of time (i.e., on the order of two hours or more) before core damage would be expected to occur if RPV injection was lost due to water carryover causing a turbine overspeed trip as the water level reaches the main steam lines. Operational experience coming out of outages when the system is drained from above the steam admission valves demonstrates that it takes less than an hour to fully drain the HPCI system and less than that amount of time to drain RCIC. This indicates that there would be more than sufficient time to drain the system prior to restarting the pumps, per the HPCI and RCIC restart procedures, if needed or desired. Therefore, even though full draining is likely not required for a successful restart, given the available procedural guidance, time available to recover from an overspeed trip, and operational experience, the failure probability assigned in the PRA for unrecoverable failure of the HPCI and RCIC turbines (0.05) is judged to be a reasonable representation for the scenarios included in the PRA model.

However, to further evaluate the uncertainty behind this assumption, sensitivity studies were run setting the assumed 0.05 failure probability for the HPCI and RCIC basic events (HPH--TTXDXI2 and RPH--TTXDXI2, respectively) to values of 0.1 and 0.5. The case of setting these basic events to 0.1 is judged to represent a realistic upper bound value sensitivity for an unrecoverable catastrophic failure of the HPCI and RCIC turbines. The case that sets these basic events to a failure probability of 0.5 is considered to be an extremely unrealistic potential upper bound value, based on the information provided above, that has the potential to mask importance measures for other PRA-modeled components. However, both cases are evaluated to support this RAI response.

The results of the sensitivity cases were evaluated for both the Full Power Internal Events (FPIE) PRA and Fire PRA (FPRA). The 0.1 sensitivity case results from the FPIE PRA and FPRA identified eleven (11) basic events as High Safety Significant (HSS) (i.e., meeting either the FV or RAW criteria for HSS as defined in NEI 00-04) which were not already HSS from the base PRA model importance results. The 0.5 sensitivity case results identified 56 basic events (including the 11 from the 0.1 sensitivity case) which were not already HSS.

The 56 basic events identified in the 0.5 sensitivity case are grouped into the following systems or types of components:

- Safety Relief Valves (SRVs), Nitrogen, and Containment Atmosphere Dilution Tank; expected because Reactor Pressure Vessel (RPV) depressurization would be required in scenarios involving HPCI and RCIC failures.
- Control Rod Drive (CRD); expected because CRD provides an alternative high-pressure injection system given sufficient time has elapsed from the plant trip.
- HPCI, RCIC and Level 8 Trip Signals; expected because these systems are an integral part of the scenarios involving potential water ingress into the steam lines.
- Independent and Dependent Operator Actions; expected because it includes actions related to the scenarios involving potential water ingress into the steam lines.

The components modeled by the additional HSS basic events identified in this sensitivity case were evaluated further to see if they would be categorized as HSS for reasons other than PRA importance values from this sensitivity study (FV or RAW). Based on this review, all of the components identified as HSS based on this PRA sensitivity would be categorized as HSS during the categorization process for other reasons. A summary of the component categorization disposition is as follows, which also includes the total number of basic events provided for each disposition category:

- The function(s) that the component(s) would be mapped to would already be HSS from the base Full Power Internal Events model or base Fire model. The components would therefore be HSS by function association. (31 basic events)
- The operator action does not change categorization of the associated components as these components would already be associated with additional independent or dependent operator actions which exceed the importance measure HSS criteria in the base Full Power Internal Events model or base Fire model results. (25 basic events)

It is important to note that the results of the 0.5 sensitivity case also resulted in a decrease in the importance measures for 73 basic events that previously exceeded the importance measures HSS criteria in the base model results. This decrease has the potential that components modeled by these basic events could be categorized as PRA LSS instead of PRA HSS. This reinforces that setting basic events HPH--TTXDXI2 and RPH--TTXDXI2 to unrealistically high failure probabilities (like 0.5) could mask importance measures of other PRA-modeled components and, therefore, impact the categorization results in a manner that is not meaningful.

This sensitivity demonstrates that even though the factor of 10 increase (use of 0.5 probability) has the potential to change the PRA basic event importance measures, there would be no net impact on the component categorization HSS results because the categorization process defined in NEI 00-04 has numerous other attributes that drive the safety significance determination. However, use of a 0.5 probability could cause a net increase in Low Safety Significant (LSS) component categorization results.

The additional basic events from the 0.5 sensitivity that would be driven HSS encompass those from the factor of 2 (0.1 probability) sensitivity, so these conclusions apply to that sensitivity as well.

ii. Alternatively, propose a mechanism that removes credit for the 0.05 failure to start probability for HPCI and RCIC turbines in the internal event and fire PRAs prior to implementation of the 10 CFR 50.69 risk informed categorization process.

Response:

Not applicable; see response to Item i above.

RAI 10.b.01 – Follow-up to Licensee Response to NRC RAI 10.b regarding External Flood Hazard Screening

In RAI 10.b, the NRC staff requested the licensee to identify any structures, systems or components (SSCs), passive and/or active, which are credited in the screening of external flood hazards. The licensee's response to RAI 10.b states that a recently issued NRC staff assessment of the Peach Bottom external flooding evaluation in a letter dated November 6, 2017 (ADAMS Accession No. ML17292B763) indicates that there are no SSCs credited in the screening of external flooding. In addition, the response states that either available physical margin exists or, where water ingress is expected, all external flood mechanisms resulted in water surface elevations below the design basis protection of the plant. However, the evaluation letter refers to "flood protection features" that are credited to reliably maintain key safety functions identified in Appendix B of NEI 16-05, Revision 1 (ADAMS Accession No. ML16165A178). These features seem to be passive features such as doors. Provide the following:

i. Clarify whether the "flood protection features" are SSCs, passive or active, that are credited in the external flood hazard screening.

ii. If the "flood protection features" are SSCs that are credited in the external flood hazard screening, then confirm that the guidance in NEI 00-04 Figure 5-6 will be followed for those SSCs (i.e. if the removal of the SSC would cause a screened scenario to become unscreened, then that SSC would be designated high-safety significant).

Response (Items i and ii):

The screening of the external flooding mechanism Local Intense Precipitation (LIP) requires permanently installed, normally closed doors (passive feature) to slow ingress of water to the Reactor Building. These doors were evaluated in the PBAPS Focused Evaluation (FE) (Reference 12). The NRC has audited the Technical Evaluation supporting the FE (Technical Evaluation 2522427-03, dated August 5, 2015 (Reference 13), and issued a Staff Assessment (Reference 14) concurring with the conclusions in the FE. Therefore, the doors credited in Technical Evaluation 2522427-03 for water retention and which are pertinent to the LIP hazard screening will be treated as HSS if categorized, in accordance with the guidance provided in NEI 00-04 Figure 5-6. The IDP will be informed of the basis for LIP screening during their reviews of categorization results.

References:

1. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Application to Adopt 10 CFR 50.69, 'Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants,'" dated August 30, 2017 (ML17243A014)

2. Letter from Richard B. Ennis (U.S. Nuclear Regulatory Commission) to Bryan C. Hanson, Exelon Generation Company, LLC – "Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Adoption of title 10 of the Code of Federal Regulations Section 50.69," dated October 10, 2017 (ML17272B016)

3. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Supplement to Application to Adopt 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," dated October 24, 2017 (ML17297B521)

4. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Draft 50.69 Request for Additional Information (RAIs) - Peach Bottom," dated March 21, 2018

5. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information - Adopt 50.69 License Amendment," dated April 6, 2018 (ML18096B506)

6. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Response to Request for Additional Information - Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," dated May 7, 2018 (ML18128A009)

7. Letter from James Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission – "Supplemental Information to Support Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants," dated June 6, 2018 (ML18157A260)

8. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information 2nd Round - Adopt 50.69 License Amendment," (Draft) dated June 26, 2018 (ML18178A477)

9. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information 2nd Round - Adopt 50.69 License Amendment," dated July 10, 2018 (ML18192A119)

10. Electronic mail message from Jennifer Tobin (U.S. Nuclear Regulatory Commission) to David Helker, Exelon Generation Company, LLC – "Peach Bottom Units 2 and 3 - Request for Additional Information 2nd Round - Adopt 50.69 License Amendment," dated July 17, 2018 (ML18200A274)

11. Kelso, J, "Nuclear Maintenance Applications Center: Terry Turbine Maintenance Guide, High-Pressure Coolant Injection (HPCI) Application," Revision 0, EPRI 3002001669, September 2013

12. D. Helker, Exelon Generation Company, LLC Response to March 12, 2012, Request for Information Enclosure 2, Recommendation 2.1, Flooding, Required Response 3, Flooding Focused Evaluation Summary Submittal, RS-17-005, March 17, 2017

13. PBAPS Technical Evaluation 2522427-03, "Assessment of LIP," August 5, 2015

14. Peach Bottom Atomic Power Station, Units 2 and 3 – Staff Assessment of Flooding Focused Evaluation (CAC Nos. MG0092 AND MG0093), dated November 6, 2017 (ML17292B763)

15. Exelon Nuclear, "High Pressure Coolant Injection (HPCI) System Recovery from System Isolation or Turbine Trip," SO 23.7.C-2, Revision 9

16. Exelon Nuclear, "Recovery from RCIC System Isolation or Turbine Trip," SO 13.7.A-2, Revision 14

ATTACHMENT 2

License Amendment Request

Peach Bottom Atomic Power Station, Units 2 and 3
Docket Nos. 50-277 and 50-278

Updated Markups of Proposed Renewed Facility Operating License (RFOL) Pages

(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall include a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the end-to-end B/Us from Peach Bottom Unit 2 benchmarking at EPU conditions. The report shall be submitted within 90 days of the completion of EPU power ascension testing for Peach Bottom Unit 3.

(f) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of the steam dryer as described in the inspection guidelines contained in WCAP-17635-P.

(g) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted within 90 days following startup from each of the first two respective refueling outages.

(h) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in paragraphs (f) and (g), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and; (2) upon satisfaction of the requirements specified in paragraph (h).

(16) <u>Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration</u>

The licensee shall not operate the facility within the MELLLA+ operating domain with a feedwater heater out of service resulting in more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

INSERT 1

3. This renewed license is subject to the following conditions for the protection of the environment:

A. To the extent matters related to thermal discharges are treated therein, operation of Peach Bottom Atomic Power Station, Unit No. 3, will be governed by NPDES Permit No. PA 0009733, as now in effect and as hereafter amended. Questions pertaining to conformance thereto shall be referred to and shall be determined by the NPDES Permit issuing or enforcement authority, as appropriate.

B. In the event of any modification of the NPDES Permit related to thermal discharges or the establishment (or amendment) of alternative effluent limitations established pursuant to Section 316 of the Federal Water Pollution Control Act, the Exelon Generation Company shall inform the NRC and analyze any associated changes in or to the Station, its components, its operation or in the discharge of effluents therefrom. If such change would entail any modification to

2. Level 1 performance criteria.

3. The methodology for establishing the RSD strain limits used for the Level 1 and Level 2 performance.

(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall include a final load definition and stress report of the steam dryer, including the results of a complete re-analysis using the end-to-end B/Us determined at EPU conditions and a comparison of predicted and measured pressures and strains (RMS levels and spectra) on the RSD. The report shall be submitted within 90 days of the completion of EPU power ascension testing for Peach Bottom Unit 2.

(f) During the first two scheduled refueling outages after reaching EPU conditions, a visual inspection shall be conducted of the steam dryer as described in the inspection guidelines contained in WCAP-17635-P.

(g) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted within 90 days following startup from each of the first two respective refueling outages.

(h) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in paragraphs (f) and (g), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and; (2) upon satisfaction of the requirements specified in paragraph (h).

(16) <u>Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration</u>

The licensee shall not operate the facility within the MELLLA+ operating domain with a feedwater heater out of service resulting in more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

Insert 1

License Amendment Request
Peach Bottom Atomic Power Station
Units 2 and 3
Adoption of 10 CFR 50.69

Updated Markups of Proposed Renewed Facility Operating License (RFOL) Pages

Unit 2 – Insert 1

(17) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

In support of implementing License Amendment No. XXX permitting the adoption of the provisions of 10 CFR 50.69 for Renewed Facility Operating License No. DPR-44 for Peach Bottom Unit 2, the license is amended to add the following license condition:

a) Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 2 License Amendment No. [XXX] dated [DATE].

   Exelon will complete the implementation items listed in Attachment 2 of Exelon's letter to the NRC dated June 6, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

   Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

Unit 3 – Insert 1

(17) Adoption of 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power plants"

In support of implementing License Amendment No. XXX permitting the adoption of the provisions of 10 CFR 50.69 for Renewed Facility Operating License No. DPR-56 for Peach Bottom Unit 3, the license is amended to add the following license condition:

a) Exelon is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards, i.e., seismic margin analysis (SMA) to evaluate seismic risk, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in Unit 3 License Amendment No. [XXX] dated [DATE].

Exelon will complete the implementation items listed in Attachment 2 of Exelon's letter to the NRC dated June 6, 2018, prior to implementation of 10 CFR 50.69. All issues identified in the attachment will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).