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

February 22, 2016

MEMORANDUM TO: Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

FROM: Richard B. Ennis, Senior Project Manager ***/RA/***
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3, DRAFT REQUEST FOR ADDITIONAL INFORMATION (CAC NOS. MF6774 AND MF6775)

The attached draft request for additional information (RAI) was transmitted on February 8, 2016, to Mr. David Neff of Exelon Generation Company, LLC (Exelon, the licensee). This information was transmitted to facilitate a conference call in order to clarify the licensee's amendment request for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, dated October 2, 2015. The proposed amendment would: (1) revise technical specification (TS) surveillance requirements (SRs) for pump flow testing of the high pressure coolant injection (HPCI) system and the reactor core isolation (RCIC) system; (2) revise the surveillance frequency requirements for verifying the sodium pentaborate enrichment of the standby liquid control (SLC) system; and (3) delete SRs associated with verifying the manual transfer capability of the normal and alternate power supplies for certain motor-operated valves associated with the suppression pool spray (SPS) and drywell spray (DWS) sub-systems of the residual heat removal (RHR) system.

The draft RAI was sent to Exelon to ensure that the questions are understandable, the regulatory basis for the questions is clear, and to determine if the information was previously docketed. During the call to discuss the draft RAI on February 22, 2016, it was determined that an RAI question from the Reactor Systems Branch was not required since the information had previously been docketed. As such, that question was deleted. In addition, during the call, the licensee agreed to provide a response to the remaining questions by March 23, 2016.

This memorandum and the attachment do not convey or represent an NRC staff position regarding the licensee's request.

Docket Nos. 50-277 and 50-278

Attachment: Draft RAI

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Docket Nos. 50-277 and 50-278

Attachment: Draft RAI

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DRAFT REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED LICENSE AMENDMENT

CHANGES TO HIGH PRESSURE COOLANT INJECTION

AND REACTOR CORE ISOLATION COOLING SURVEILLANCE REQUIREMENTS

EXELON GENERATION COMPANY, LLC

PEACH BOTTOM ATOMIC POWER STATION - UNITS 2 AND 3

DOCKET NOS. 50-277 AND 50-278

By letter dated October 2, 2015 (ADAMS Accession No. ML15275A265), Exelon Generation Company, LLC (Exelon, the licensee) submitted a license amendment request (LAR) for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed amendment would: (1) revise technical specification (TS) surveillance requirements (SRs) for pump flow testing of the high pressure coolant injection (HPCI) system and the reactor core isolation (RCIC) system; (2) revise the surveillance frequency requirements for verifying the sodium pentaborate enrichment of the standby liquid control (SLC) system; and (3) delete SRs associated with verifying the manual transfer capability of the normal and alternate power supplies for certain motor-operated valves associated with the suppression pool spray (SPS) and drywell spray (DWS) sub-systems of the residual heat removal (RHR) system.

The Nuclear Regulatory Commission (NRC) staff has reviewed the information the licensee provided that supports the proposed amendment and would like to discuss the following issues to clarify the submittal.

Containment and Ventilation Branch (SCVB)
Reviewer: Ahsan Sallman

SCVB-RAI-1

The SPS and the DWS are safety-related subsystems or operating modes of the RHR system which perform containment cooling function during a design basis accident. These modes have associated Limiting Conditions of Operation (LCOs) with SRs. In the licensee's letter dated October, 2, 2015, Attachment 1, Section 3.0, last paragraph under heading "SR 3.6.2.4.3 and SR 3.6.2.5.3 Deletions" provides the following justification for deleting SR 3.6.2.4.3 and SR 3.6.2.5.3 for the LCOs for these RHR modes:

> While the use of the alternate power supply could be used for operation of the SPS and DWS sub-systems, the need for this capability would involve more than a single failure. The alternate power supply capability for the specific RHR valves is not relied upon for the design function of the SPS and DWS subsystems. The accident analyses do not rely on the use of the alternate power supply for the SPS and DWS sub-systems assuming any single failure. The License Amendment Request for the PBAPS EPU [Extended Power Uprate]

> misinterpreted the FMEA [Failure Modes and Effects Analysis] to rely on the use of the alternate power supply for all modes of containment cooling instead of only the SPC function.

a) The above statement does not clearly justify deletion of SR 3.6.2.4.3 and SR 3.6.2.5.3 for the SPS and DWS modes while similar SR 3.6.2.3.3 for the RHR Suppression Pool Cooling (SPC) mode is required, even though all three (SPC, SPS, and DWS) are safety-related modes of RHR system credited for containment cooling function during a design basis accident. Describe the safety design basis functions of the SPS and DWS modes and further explanation and reasons for deleting the SRs.

b) Please explain why the need for the alternate power supply capability for the SPS and DWS modes would involve more than a single failure. Describe the possible failures that would require the need for SPS and DWS to perform their safety design basis functions.

SCVB-RAI-2

Is SPS and DWS credited for containment atmosphere cleanup as required by 10 CFR Part 50 Appendix A, General Design Criteria (GDC)-41[1], which requires systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents?

a) In case SPS and DWS are not credited, please explain which systems are provided to meet the AEC draft GDCs equivalent to current GDC-41.

b) In case the SPS and DWS are credited for fission product cleanup, the RHR control valve and cross-tie valves must be used for achieving a positive net positive suction head (NPSH) margin for the RHR pumps without using containment accident pressure (CAP). Please explain why an alternate safety-related power source for these valves would not be needed in case of loss of the normal safety-related power source.

SCVB-RAI-3

Section 4.3 of Attachment 1 to the licensee's letter dated October 2, 2015, states, in part, that:

> These specific changes delete the SR 3.6.2.4.3 from the Suppression Pool Spray (SPS) TS section and delete SR 3.6.2.5.3 from the Drywell Spray (DWS) TS section since the alternate power supply capability is not necessary for the SPS or DWS sub-systems of the RHR system to perform their design functions.

Please explain why alternate power supply capability is not necessary for the SPS or DWS sub-systems of the RHR system to perform their design functions.

SCVB-RAI-4

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1 The PBAPS evaluation of the comparable 1967 Atomic Energy Commission (AEC) proposed General Design Criteria (referred to here as "Draft GDC") is contained in PBAPS UFSAR Appendix H, Draft GDC-11, Draft GDC-17, Draft GDC-69, and Draft GDC-70.

Section 4.3 of Attachment 1 to the licensee's letter dated October 2, 2015, states, in part, that:

> …deletion of the SPS and DWS sub-system SRs 3.6.2.4.3 and 3.6.2.5.3 do not affect the ability of these systems to perform their design functions. These proposed changes are administrative in nature and have no effect on plant operation.

Please explain why the deletion of SRs 3.6.2.4.3 and 3.6.2.5.3 is considered as administrative in nature.

SCVB-RAI-5

As discussed in Reference 3, Section 2.6.1.2.1, under the heading "Loss-of-Coolant Accident Loads", the Emergency Operating Procedures (EOPs) require DWS prior to Wetwell (WW) pressure exceeding 9 psig as stated below:

> However, emergency operating procedures (EOPs) for PBAPS include direction to initiate DW sprays prior to WW pressure exceeding 9.0 psig. Containment analyses performed for PBAPS EPU have shown that WW [Wetwell] pressure will exceed this DW [Drywell] spray initiation pressure of 9.0 psig before 900 seconds following initiation of the event. <u>Initiation of DW sprays will rapidly reduce DW pressure and stop chugging</u>. [emphasis added]

As discussed in Reference 3, Section 2.6.5.1, under the heading "Pool Temperature Response – RSLB [recirculation Suction Line Break] DBLOCA," the shutdown cooling of the non-accident unit under a dual unit interaction uses the DWS and SWS spray modes to maintain the containment shell temperature less than the shell design temperature as stated below:

> The non-accident unit containment cooling interruption is assumed for a period of 10-minutes due to the dual unit interaction. After the 10-minute interruption, containment cooling on the non-accident unit is restored with the same containment cooling configuration as existed prior to the dual unit interaction interruption. When RPV pressure reaches 150 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Containment cooling is maintained using RHR SPC mode. <u>The containment spray mode of RHR may be initiated to maintain the containment shell temperature less than the containment design temperature of 281°F.</u> [emphasis added]

As discussed in Reference 3, Section 2.6.5.1, under the heading "Pool Temperature Response – Small Steam Break LOCA," for single unit analysis, it states that:

> At 10 minutes, operators turn off two of the RHR pumps, and align the remaining RHR pump to provide containment cooling with a flow of 8600 gpm through one RHR heat exchanger, and one HPSW pump providing cooling flow to the RHR heat exchanger. When DW pressure exceeds 2.0 psig, operators initiate WW spray, with the remainder of the RHR flow remaining in SPC mode. <u>Prior to WW pressure exceeding 9.0 psig, operators initiate DW spray and stop all SPC flow</u>. When bulk suppression pool temperature exceeds 110°F (but not earlier than 10

> minutes following initiation of the event), operators initiate a controlled reactor vessel cooldown at 100°F per hour. At one hour from initiation of the event, operators establish the RHR heat exchanger cross-tie to the other RHR heat exchanger in the same loop, such that a total RHR flow rate of 8600 gpm is maintained to the DW and WW spray headers. When reactor pressure is decreased below the pressure permissive for NSDC (70 psig), operators maintain DW and WW spray cooling with the RHR heat exchanger cross-tie in service, and maintain reactor vessel pressure as low as possible to limit steam flow from the break. When bulk suppression pool temperature is below a pre-determined value (170°F for EPU), operators open one or more ADS valves and increase vessel water level in the reactor vessel to the MSL nozzles using the one loop of CS until water flows from the open ADS valves back to the suppression pool, establishing ASDC. DW and WW sprays continue to be used to provide containment and SPC, bulk reactor water temperature decreases below 200°F and cold shutdown is achieved prior to 36 hours from initiation of the event, which is conservative for plants with cold shutdown defined at a higher temperature. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 2.6-11 and the peak bulk suppression pool temperature at 187°F. [emphasis added]

As discussed in Reference 3, Section 2.6.5.1, under the heading "Pool Temperature Response – Small Steam Break LOCA," for dual unit interaction analysis, it states that:

> At 10 minutes, operators turn off two of the RHR pumps and align the remaining RHR pump to provide containment cooling using with a flow of 8600 gpm through one RHR heat exchanger and one HPSW pump providing cooling flow to the RHR heat exchanger. When DW pressure exceeds 2.0 psig, operators initiate WW spray, with the remainder of the RHR flow remaining in SPC mode. Prior to WW pressure exceeding 9.0 psig, operators initiate DW spray and stop all SPC flow**.** When bulk suppression pool temperature exceeds 110°F (but not earlier than 10 minutes following initiation of the event), operators initiate a controlled reactor vessel cooldown at 100°F per hour. At one hour from initiation of the event, operators establish the RHR heat exchanger cross-tie to the other RHR heat exchanger in the same loop, such that a total RHR flow rate of 8600 gpm is maintained to the DW and WW spray headers**.** At one-hour following the start of reactor depressurization it is assumed that a LOCA signal on the SSLB LOCA unit may occur on HDWP commensurate with low reactor pressure. This timing for the HDWP/low reactor pressure LOCA signal on the small break LOCA/accident unit is based on the depressurization rate of 100°F/hr mentioned above. [emphasis added]

As stated in the underlined portions of the above Sections of Reference 3, the large break LOCA and the Small Steam Break LOCA containment cooling analysis credit the RHR DWS and SPS modes. In case these modes are not available, due to loss of normal power source to the RHR cross tie and control valves, describe the alternate safety system for reducing chugging, mitigating large and small steam break LOCAs, and safe shutdown cooling of the non-accident unit that should be credited in the analysis,

REFERENCES

1. Exelon letter to NRC dated October 2, 2015, "License Amendment Request Proposed Changes to the Technical Specifications for High Pressure Coolant Injection and Reactor Core Isolation Cooling Surveillance Test Pressure and Clarification of Surveillance Requirements," (ADAMS Accession No. ML15275A265).

2. Exelon letter to NRC dated September 28, 2012, "License Amendment Request - Extended Power Uprate", (ADAMS Accession No. ML12286A012).

3. General Electric-Hitachi Report NEDC-33566, Revision 0, Attachment 6 to Exelon letter to NRC dated September 28, 2012 (Reference 2), "NEDC-33566P, Safety Analysis Report for Exelon Peach Bottom Atomic Power Stations Units 2 and 3 Constant Pressure Power Uprate," (Non-Public, Proprietary).