Exelon Generation.

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10 CFR 50.90

December 06, 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
License Amendment Request - Revise Technical Specifications to Allow Two Safety Relief Valves/Safety Valves to be Out-of-Service with Increased Reactor Pressure Safety Limit

References:
1. Exelon Letter to the NRC, "License Amendment Request – Revise Technical Specifications to Allow Two Safety Relief Valves/Safety Valves to be Out-of-Service with Increased Reactor Pressure Safety Limit," dated May 30, 2018 (ADAMS Accession No. ML18150A387)

2. NRC Email to Exelon, "Peach Bottom Units 2 and 3 - Request for Additional Information (nonpublic)- LAR to Allow 2SRV/SVs OOS at High Pressure (EPID L-2018-LLA-0151)," dated November 15, 2018 (ADAMS Accession No. No. ML18324A674)

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon), proposed a change to the Technical Specifications (TSs), Appendix A of Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, on May 30, 2018 to allow continued operation with two Safety Relief Valves/Safety Valves (SRVs/SVs) out-of-service with Increased Reactor Pressure Safety Limit (Reference 1).

During their technical review of the application, the NRC Staff identified the need for additional information. Reference 2 provided the Request for Additional Information (RAI) from the Reactor Systems Branch. A response to the RAI was requested to be submitted by December 10, 2018. The Attachment to this letter provides the response to this RAI.

Exelon has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the U.S. Nuclear Regulatory Commission in Reference 1. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. Further, the additional information provided in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed amendment.

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

There are no regulatory commitments contained in this response.

Should you have any questions concerning this letter, please contact Mr. David Neff at (267) 533-1132.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 06th day of December 2018.

Respectfully,

David T. Gudger

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

Attachment: Response to Request for Additional Information from NRC Review Branch SRXB

cc:	Regional Administrator - NRC Region I	w/ Attachment
	NRC Senior Resident Inspector - PBAPS	"
	NRC Project Manager, NRR - PBAPS	"
	R. R. Janati, Pennsylvania Bureau of Radiation Protection	"
	D. A. Tancabel, State of Maryland	"

ATTACHMENT

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 from NRC Review Branch SRXB
License Amendment Request - Revise Technical Specifications to Allow
Two Safety Relief Valves/Safety Valves to be Out-of-Service
with Increased Reactor Pressure Safety Limit

Response to NRC Staff's
Request for Additional Information

By application, dated May 30, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18150A387), Exelon Generating Company, LLC submitted a License Amendment Request (LAR) for Peach Bottom Atomic Power Station, Units 2 and 3 (PBAPS). The proposed LAR would revise PBAPS Technical Specifications to allow continued operation with two Safety Relief Valves/Safety Valves (SRVs/SVs) out-of-service and to increase the Reactor Coolant System Pressure Safety Limit (SL).

In an email dated November 15, 2018, from the NRC (Jennifer Tobin) to Exelon (David Helker) (ADAMS Accession No. ML18324A674), the NRC provided a Request for Additional Information (RAI) seeking clarification of certain issues related to the submittal. A response to the RAIs was requested to be submitted by December 10, 2018.

RAI-SRXB-1: ASME Overpressure Analysis with New Reactor Pressure Limit

Draft GDCs 9, 33 and final GDC 31 require overpressure protection during power operation be provided by relief/safety valves (SRVs/SVs) and protection system. The LAR proposed to raise a new reactor coolant system pressure safety limit so that the impact of the ASME overpressure analysis with 2 SRVOOS can be accepted. To facilitate the staff review, provide the following information associated with the analysis as provided in the LAR:

1. Peach Bottom technical specification bases 2.1.2 indicates the RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes. Please verify the locations for the peak vessel pressure as reported in Tables 1 and 2 of the LAR are consistent with the TS bases.

   RESPONSE

   The PBAPS Technical Specification (TS) Bases Section 2.1.2 indicates that the Reactor Coolant System (RCS) pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The bases further identify the American Society of Mechanical Engineer (ASME) Pressure Vessel Code limit of 1375 psig as the acceptance criterion for system integrity. The TS Bases Section 2.1.2 also indicate that the specified SL, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. As with the standard Boiling Water Reactor (BWR) design, the PBAPS design does not measure reactor pressure at the lowest elevation of the RCS, but rather in the reactor steam dome. Accordingly, the steam dome pressure is utilized as a surrogate for peak vessel pressure in the TS Section 2.1.2. Because of the natural pressure differential between the steam dome and the lowest elevation of the RCS, the steam dome SL value must be lower than the ASME Pressure Vessel Code limit of 1375 psig. The design pressure of the RCS vessel (1250 psig) is bounded by the design pressure of the RCS piping. Thus, the RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes. The peak vessel pressure values reported in Tables 1 and 2 of Attachment 1 of the LAR represent the pressure in the lowest RCS volume of the TRACG-AOO model. The elevation of the vertical midpoint of this volume is 0.65 meters (2.1 ft) above the bottom of the vessel. The static elevation head of this additional 2.1 feet of elevation above the bottom of the RCS is less than 1 psi, which is insignificant. The peak vessel pressure values reported in Tables 1 and 2 of Attachment 1 of the LAR are therefore consistent with the TS Bases Section 2.1.2.

2. A verification of whether a TRACG statistical pressure adder had been applied to the peak vessel pressure as reported in the Tables 1 and 2 of Attachment 1 of the LAR. Note that it is known that an adder will be applied to the peak steam dome pressure. However, it is not clear if an adder will also be applied to the peak vessel pressure to be reported. Provide justification if the TRACG statistical pressure adder is not applied.

   RESPONSE

   The TRACG statistical pressure adder was applied to the peak vessel pressure as reported in the Tables 1 and 2 of the LAR, Attachment 1.

3. Justify that if the steam dome pressure were to approach the proposed reactor steam dome limit of 1340 psig the corresponding peak vessel pressure will still be below the ASME limit of 1375 psig with margin.

   RESPONSE

   Analysis experience for overpressure analysis has identified that the differential pressure between the peak dome pressure and peak vessel pressure is relatively constant for a given analysis statepoint (e.g., Increased Core Flow (ICF), Maximum Extended Load Line Limit Analysis Plus (MELLLA+)). This is illustrated in the analysis results presented in Tables 1 and 2 of Attachment 1 of the LAR. Examination of the ICF cases for both Unit 2 and Unit 3, and for both 1 SRV/SV out-of-service (SRVOOS) and 2 SRVOOS, identifies that the differential peak pressure between the steam dome and vessel (lowest point) is between 29 and 30 psi for all cases. Similarly, for the MELLLA+ cases, the differential peak pressure between the steam dome and vessel is between 24 and 25 psi for all cases.

   The differential peak pressure is a function of the reactor hydraulic characteristics and overpressure event sequence/progression, both of which remain essentially the same from cycle to cycle. This is further corroborated by examination of the PBAPS Unit 2 Cycle 22 pre-Measurement Uncertainty Recapture Uprate (MUR) results for the limiting ICF case, which results in a peak steam dome pressure of 1320 psig and a peak pressure differential of 29 psi; the PBAPS Unit 3 Cycle 21 pre-MUR results for the limiting ICF case, which results in a peak steam dome pressure of 1318 psig and a peak pressure differential of 29 psi; and the PBAPS Unit 2 Cycle 21 pre-MUR results for the limiting ICF case, which results in a peak steam dome pressure of 1313 psig and a peak pressure differential of 28 psi.

   The limiting case for the peak vessel pressure (Table 1 of Attachment 1 of the LAR, ICF (HBB), 2 SRVOOS) results in a peak steam dome pressure of 1325 psig and a peak vessel pressure of 1354 psig. For this case, the margin to the proposed steam dome SL (1340 psig) is 15 psi and the margin to the ASME code pressure limit (1375 psig) is 21 psi. Given that the peak differential pressure is expected to remain essentially the same, a case that would result in the peak steam dome pressure of 1340 psig (the SL) would result in a peak vessel pressure approximately 6 psi less than the ASME code pressure limit. This result is illustrated in Table 3 of Attachment 1 of the LAR and is discussed in the associated text. Because the calculated peak vessel pressure has more margin to the ASME limit (1375) than the calculated peak steam dome pressure has to the proposed TS SL (1340 psig), the analysis results will always reach the TS steam dome SL limit before the RCS pressure would reach the ASME peak vessel limit, thus ensuring compliance with the ASME limit. Furthermore, the reload analysis process and the associated cycle specific Supplemental Reload Licensing Report (SRLR) evaluate/report both the steam dome results and the peak vessel results to verify/validate that neither the TS SL nor the ASME code limit is exceeded.