From:	Tobin, Jennifer
To:	"Helker, David P:(GenCo-Nuc)"
Cc:	"Gropp Jr, Richard W:(GenCo-Nuc)"; Danna, James
Subject:	Peach Bottom Units 2 and 3 - Request for Additional Information - High Pressure Service Water One Time TS Change (EPID L-2018-LLA-0265)
Date:	Wednesday, January 16, 2019 11:52:00 AM

Dear Mr. Helker,

By letter dated September 28, 2018 (Accession No. ML18275A023), Exelon Generation Company, LLC requested to change technical specifications for Peach Bottom Atomic Power Station Units 2 and 3. The proposed change would revise the design and licensing basis described in the final safety analysis report (as updated) to reduce the design pressure rating of the high pressure service water system.

The Nuclear Regulatory Commission's (NRC) staff is reviewing your submittal and has determined that additional information is needed to complete its review. The specific request for additional information (RAI) questions are provided below. These questions are being sent to ensure that the questions are understandable, the regulatory basis for the questions is clear, and to determine if the information was previously docketed. A clarification phone call to discuss the RAIs was held January 16, 2019. No changes were made to the draft RAIs as a result of the call. A response to these RAIs is requested by February 18, 2019.

If you have any questions, please contact me at (301) 415-2328. A copy of this email with the RAIs will be made publicly available in ADAMS.

Thanks,
Jenny

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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR A LICENSE AMENDMENT REQUEST TO REDUCE HIGH PRESSURE SERVICE WATER SYSTEM DESIGN PRESSURE AND REVISE TECHNICAL SPECIFICATIONS 3.6.2.3, 3.6.2.4, 3.6.2.5, AND 3.7.1 FOR TEMPORARY EXTENSION OF COMPLETION TIMES

EXELON GENERATION COMPANY LLC

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

DOCKET NUMBERS 50-277 AND 50-278

ENTERPRISE PROJECT IDENTIFIER L-2018-LLA-0265

By letter dated September 28, 2018, Exelon Generation Company LLC, the licensee, proposes to change technical specifications for Peach Bottom Atomic Power Station Units 2 and 3. The proposed change would revise the design and licensing basis described in the final safety analysis report (as updated) to reduce the design pressure rating of the high pressure service water system. In addition, the proposed change includes a change to technical specifications (TS) 3.6.2.3, “Residual Heat Removal (RHR) Suppression Pool Cooling,” TS 3.6.2.4, “Residual Heat Removal (RHR) Suppression Pool Spray,” TS 3.6.2.5, “Residual Heat Removal (RHR) Drywell Spray,” and TS 3.7.1, “High Pressure Service Water (HPSW) System,” completion times.

During the Nuclear Regulatory Commission (NRC) staff’s review of the license amendment request, the NRC staff determined that more information was needed to complete the review.

Regulatory Analysis Basis

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.67, “Accident Source Term,” allows licensees to revise their current accident source term in the design basis radiological consequence analyses to apply for a license amendment under § 50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report. Section 50.67(b)(2) requires that the licensee's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

Background

In the license amendment request the licensee proposes to modify the HPSW system design, which will result in a reduction of the design pressure rating of the HPSW system to a value of approximately 200 psig. At this design pressure the HPSW system, when in operation, will no longer be maintained at a higher pressure than the RHR system. The RHR system injects water into the reactor vessel following a loss of coolant accident. After the proposed change to the HPSW system, if there is leakage at the RHR heat exchanger interface, during a loss of coolant accident this would allow the transport of radioactive

material into the HPSW system and ultimately to the environment via the discharge canal.

In the licensee amendment request the licensee stated:

> No changes are required to the UFSAR [updated final safety analysis report] Chapter 14 accident analyses. The calculated radiological consequences of analyzed accidents are unchanged because the design basis radiological analysis, as provided in Regulatory Guide 1.183, does not consider any liquid effluent release pathway through a heat exchanger interface. No new release pathway is introduced with the proposed change with respect to the design basis radiological analysis.
>
> Though not required per the station design basis, a technical evaluation was performed to consider the radiological impact of a post-LOCA [loss of coolant accident] plant release into the environment via a liquid release pathway from a RHR heat exchanger leaking into HPSW. There is no regulatory guidance for a post-LOCA liquid release pathway and it is considered to be outside of the design bases. The most limiting dose consequence from RHR heat exchanger leakage into HPSW is that associated with post-LOCA control room dose due to airborne iodine release fraction. Based on a conservative analysis and maximum allowable RHR heat exchanger leak rate, the post-LOCA control room 30-day dose remains below the regulatory limit (5 rem TEDE). In addition, the evaluation considered the dose consequence from RHR heat exchanger leakage into HPSW at the nearest potable water sources and concluded that the 30-day dose to the public remains below the 25 rem TEDE limit at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ).
>
> An evaluation was also performed to consider the radiological impact due to a potential increase in RHR heat exchanger leakage during normal operations resulting from periods of operation in which HPSW pressure may be less than RHR pressure. A parametric study was conducted, using conservative RHR heat exchanger leakage rates, and concluded that the Offsite Dose Calculation Manual (ODCM) limits would not be exceeded.

Regulatory guide 1.183 revision 0, Appendix A, regulatory position 5 states:

> ESF [engineered safety features] systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components … The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs [boiling water reactors] and PWRs [pressurized water reactors].

The intent of regulatory position 5 of regulatory guide 1.183 is to analyze the radiological

consequences for all possible leakage pathways from ESF systems. The ESF leakage pathways stated in regulatory position 5 is not a complete list of all possible pathways. For Peach Bottom Atomic Power Station, RHR heat exchanger leakage was not included in the loss of coolant accident radiological consequence analysis because the HPSW system was operated at a higher pressure than the RHR system thus preventing any outward leakage from the RHR system; because of this, it wasn't necessary to evaluate this RHR leakage pathway. However, the proposed change no longer prevents RHR leakage through the heat exchanger interface to the HPSW system and it allows any RHR leakage to reach the environment via the HPSW system to the discharge canal.

In order for the NRC staff to be able to state that there is reasonable assurance that the regulatory limits stated in 10 CFR 50.67 continue to be met at Peach Bottom Atomic Power Station for the LOCA, the licensee's evaluation of the dose consequences from RHR leakage through the RHR heat exchanger interface must be reviewed.

RAI-1

Therefore, please provide additional information describing the assumptions, inputs, methodology, and results of the aforementioned RHR leakage dose consequence analysis.

RAI-2

In addition please provide the basis for why any increase in the calculated dose from this new pathway (previously precluded by pressure differential) should not be included in the design basis LOCA analysis of record.