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10 CFR 50.90

March 15, 2016

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 regarding License Amendment Request to Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits

References:
1) License Amendment Request to Reduce Steam Dome Pressure Specified in the Reactor Core Safety Limits dated December 15, 2015
2) Email correspondence from R. Ennis (U.S. Nuclear Regulatory Commission) to S. Hanson (Exelon Generation Company, LLC), "Peach Bottom Atomic Power Station, Units 2 and 3 – DRAFT Request for Additional Information regarding Proposed License Amendment to Reduce Steam Dome Pressure Specified in Reactor Core Safety Limits (CAC NOS. MF7184 & MF7185)," dated February 18, 2016

On December 15, 2015, in accordance with the provisions of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon), submitted a request for an amendment to the Technical Specifications (TS), Appendix A of Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3, respectively (Reference 1).

The proposed amendment requested U.S. Nuclear Regulatory Commission (NRC) approval to reduce the reactor steam dome pressure specified in Technical Specification (TS) 2.1.1 for the reactor core safety limits. The proposed change addresses a Title 10 of the Code of Federal Regulations (10 CFR), Part 21 issue concerning the potential to violate the safety limits during a pressure regulator failure maximum demand (open) (PRFO) transient.

The NRC reviewed the license amendment request and identified the need for additional information in order to complete its evaluation of the amendment request. The draft request for additional information (RAI) was sent from the NRC to Exelon by electronic mail message on February 18, 2016 (Reference 2).

Our response to the draft RAI is provided in Attachment 1. Attachment 2 provides revised TS markups. For information, Attachment 3 provides revised TS Bases markups.

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 supplement 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 supplement 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 commitments contained within 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 supplement 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 Stephanie J. Hanson at 610-765-5143.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of March 2016.

Respectfully,

James Barstow

James Barstow
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1. Response to Request for Additional Information
2. Revised Markup of Proposed Technical Specifications Pages
3. Revised Markup of Proposed Technical Specifications Bases Pages (For Information Only)

cc:	Regional Administrator - NRC Region I	w/attachments
	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	"
	S. T. Gray, State of Maryland	"

ATTACHMENT 1

Response to Request for Additional Information Regarding License Amendment Request to Reduce Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

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

By letter to the Nuclear Regulatory Commission (NRC) dated December 15, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15349A800), 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 reduce the reactor steam dome pressure stated in Technical Specification (TS) 2.1.1 for the reactor core safety limits. The proposed change addresses a Title 10 of the Code of Federal Regulations (10 CFR), Part 21 issue concerning the potential to violate the safety limits during a pressure regulator failure maximum demand (open) (PRFO) transient. The question is restated below along with Exelon's response.

DORL-RAI-1

The licensee proposes to reduce the reactor steam dome pressure specified in TS 2.1.1.1 and TS 2.1.1.2 from 785 pounds per square inch gauge (psig) to 685 psig based on the lower-bound pressure for the critical power correlation for the GNF2 fuel currently used in the PBAPS, Units 2 and 3 cores. The licensee's application references Global Nuclear Fuel (GNF) reports NEDC-33270P and NEDC-33292P as the basis supporting the proposed change.

Section 3.8.3 of GNF report NEDC-33270P discusses the critical power correlation for GNF2 fuel (i.e., GEXL17 correlation). This section includes the pressure range over which the GEXL17 correlation is valid for GNF2 fuel consistent with the information provided in Table 5-4 of GNF2 report NEDC-33292P. As discussed in Section 3.0 of Attachment 1 of the licensee's application, the lower bound pressure limit for the GEXL17 correlation is 700 pounds per square inch atmospheric (psia). Converting to psig, this lower bound pressure is approximately 685.3 psig. As such, the 685 psig value specified in the proposed TS change is slightly outside the pressure range in which the GEXL17 correlation is valid for GNF2 fuel. Please provide further justification for the proposed 685 psig value or propose a revised pressure value for this TS change that is supported by the GEXL17 correlation.

RESPONSE

This request for additional information was discussed during a conference call with the NRC on February 23, 2016. Based on follow-up discussions with the NRC, Exelon has decided to reference the lower bound limit for the critical power correlation in absolute pressure (i.e., 700 psia) for the GNF2 fuel currently used in the PBAPS, Units 2 and 3 cores, as referenced by the Global Nuclear Fuel (GNF) reports, NEDC-33270P and NEDC-33292P.

Exelon proposes to revise the lower bound pressure for the reactor core safety limits specified in TS 2.1.1.1 and TS 2.1.1.2 to reference the absolute pressure value of 700 psia.

Attachment 2 provides a copy of the revised marked up TS pages that reflect the proposed change. For information, a copy of the revised marked up TS Bases pages that reflect the proposed change are provided in Attachment 3.

ATTACHMENT 2

Revised Markup of Technical Specifications Pages

Response to Request for Additional Information Regarding License Amendment Request to Reduce Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

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

Unit 2 TS Pages
2.0-1

Unit 3 TS Pages
2.0-1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785 psig~~ 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785 psig~~ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.14 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785 psig~~ 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785 psig~~ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

(continued)

ATTACHMENT 3

Revised Markup of Technical Specifications Bases Pages

Response to Request for Additional Information Regarding License Amendment Request to Reduce Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

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

<u>Unit 2 TS Bases Pages</u>
B 2.0-2
B 2.0-3
B 2.0-6
B 3.3-147

<u>Unit 3 TS Bases Pages</u>
B 2.0-2
B 2.0-3
B 2.0-6
B 3.3-148

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormal operational transients. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

<u>2.1.1.1 Fuel Cladding Integrity</u>

700 psia

GE critical power correlations are applicable for all critical power calculations at pressures ≥ ~~785 psig~~ and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> The pressure drop in the bypass region is essentially all elevation head with a value > 4.5 psi; therefore, the core pressure drop at low power and flows will always be > 4.5 psi. At power, the static head inside

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the bundle is less than the static head in the bypass region because the addition of heat reduces the density of the water. At the same time, dynamic head loss in the bundle will be greater than in the bypass region because of two phase flow effects. Analyses show that this combination of effects causes bundle pressure drop to be nearly independent of bundle power when bundle flow is 28 X 10^3 lb/hr and bundle pressure drop is 3.5 psi. Because core pressure drop at low power and flows will always be > 4.5 psi, the bundle flow will be > 28 X 10^3 lb/hr.

Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power with bundle flow at 28 X 10^3 lb/hr is approximately 3.35 MWt. This is equivalent to a THERMAL POWER > 50% RTP even when design peaking factors are considered. Therefore, a THERMAL POWER limit of 23% RTP ~~prevents any bundle from exceeding critical power and is a conservative limit when reactor pressure < 785 psig.~~

for reactor pressure < 700 psia is conservative. Additional information on low flow conditions is available in Reference 4.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore,

(continued)

BASES

REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," latest approved revision.

2. 10 CFR 100.

3. 10 CFR 50.67.

4. SIL No. 516 Supplement 2, January 19, 1996.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level—Low Low Low (Level 1) (continued)

The Reactor Vessel Water Level—Low Low Low (Level 1) Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

1.b. Main Steam Line Pressure—Low

during the depressurization transient in order to maintain reactor steam dome pressure > 700 psia. The MSIV closure

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure—Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 3). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs ~~prior to pressure decreasing below 785 psig, which~~ results in a scram ~~due to MSIV closure~~, thus reducing reactor power to < 23% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure—Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure—Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 1).

This Function isolates MSIVs, MSL drains, MSL sample lines and recirculation loop sample line valves.

(continued)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

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700 psia

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