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10 CFR 50
10 CFR 51
10 CFR 54

March 18, 2019

U.S. Nuclear Regulatory Commission
Attention: 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: Supplement No. 4 - Changes to the Peach Bottom Atomic Power Station, Units 2 and 3, Subsequent License Renewal Application

References:

1. Letter from Michael P. Gallagher, Exelon Generation Company, LLC (Exelon) to NRC Document Control Desk, dated July 10, 2018, "Application for Subsequent Renewed Operating Licenses"

2. Letter from Michael P. Gallagher, Exelon Generation Company, LLC (Exelon) to NRC Document Control Desk, dated September 14, 2018, "Changes to the Peach Bottom Atomic Power Station, Units 2 and 3, Subsequent License Renewal Application" (Supplement No. 1)

3. Letter from Michael P. Gallagher, Exelon Generation Company, LLC (Exelon) to NRC Document Control Desk, dated January 23, 2019, "Changes to the Peach Bottom Atomic Power Station, Units 2 and 3, Subsequent License Renewal Application" (Supplement No. 2)

4. Letter from Michael P. Gallagher, Exelon Generation Company, LLC (Exelon) to NRC Document Control Desk, dated February 11, 2019, "Changes to the Peach Bottom Atomic Power Station, Units 2 and 3, Subsequent License Renewal Application" (Supplement No. 3)

In Reference 1, Exelon submitted the Subsequent License Renewal Application (SLRA) for the Peach Bottom Atomic Power Station, Units 2 and 3 (PBAPS). In References 2, 3, and 4, Exelon submitted Supplement Nos. 1, 2, and 3 to the SLRA for PBAPS. The purpose of this letter is to provide Supplement No. 4 to the SLRA for PBAPS. Supplement No. 4 includes one change to the SLRA which provides additional information and clarification in the SLRA to address the NRC Safety Review Audit information needs.

Enclosure A to this letter provides a description of the change, and corresponding mark-ups to the affected portion of the SLRA, thereby supplementing the PBAPS SLRA.

This letter contains no new regulatory commitments.

This submittal has been discussed with the NRC License Renewal Senior Project Manager for the PBAPS Subsequent License Renewal project.

If you have any questions, please contact Mr. David Distel, Licensing Lead, Exelon License Renewal Projects, at 610-765-5517.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of March 2019.

Respectfully submitted,

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Exelon Generation Company, LLC

Enclosure: A. Changes to the PBAPS Subsequent License Renewal Application

cc: Regional Administrator – NRC Region I
NRC Senior Project Manager (Safety Review), NRR-DMLR
NRC Project Manager (Environmental Review), NRR-DMLR
NRC Project Manager, NRR-DORL – Peach Bottom Atomic Power Station
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Enclosure A

Changes to the PBAPS Subsequent License Renewal Application

Introduction

This Enclosure contains one change that is being made to the Subsequent License Renewal Application (SLRA) that was identified after submittal of the SLRA. For this item, the change is described and the affected page number(s) and portion(s) of the SLRA is provided. For clarity, entire sentences or paragraphs from the SLRA are provided with deleted text highlighted by ~~strikethroughs~~ and inserted text highlighted by ***bolded italics***.

Revision of SLRA Section 3.5.2.2.2.6 for irradiation of steel supports and structures

Affected SLRA Sections: Section 3.5.2.2.2.6

SLRA Page Numbers: 3.5-54 and 3.5-56

Description of Change:

In addition to considering the potential aging effects due to irradiation of reinforced concrete in SLRA Section 3.5.2.2.2.6, during recent SLR audits, the NRC identified that a reduction in fracture toughness due to irradiation embrittlement of the reactor vessel support steel and the sacrificial shield wall steel elements is a potential aging effect that should be addressed. The further evaluation in SLRA Section 3.5.2.2.2.6 is revised to address the potential aging effects to these steel components due to irradiation. The revision to the further evaluation provides the basis for concluding that the aging effects due to irradiation are not significant for the reactor vessel support skirt and steel components of the sacrificial shield wall. Therefore, a plant specific program is not required to manage the aging effects due to irradiation of the reactor vessel support skirt and steel components of the sacrificial shield wall. An additional correction was made to SLRA Section 3.5.2.2.2.6 to correct the gamma dose to the sacrificial shield wall concrete and a reference number that was incorrect.

Accordingly, SLRA Section 3.5.2.2.2.6 is revised.

SLRA Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation is revised as shown below. For clarity, the entire further evaluation is being shown. The changes include those shown at the bottom paragraph on page 3-54, with the correction of the reference to Table 4.2.1.1-1, and on page 3.5-56, with a correction of the estimated gamma dose in the next to last paragraph of the original section. New subsections are added to the end of SLRA Section 3.5.2.2.2.6 to address irradiation of the reactor vessel support and the sacrificial shield wall. There is no change to the rest of the Section 3.5.2.2.2.6, which addresses irradiation of concrete.

3.5.2.2.2.6 Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation

Reduction of strength, loss of mechanical properties, and cracking due to irradiation could occur in PWR and BWR Group 4 concrete structures that are exposed to high levels of neutron and gamma radiation. These structures include the reactor (primary/biological) shield wall, the sacrificial shield wall, and the reactor vessel support/pedestal structure. Data related to the effects and significance of neutron and gamma radiation on concrete mechanical and physical properties is limited, especially for conditions (dose, temperature, etc.) representative of light-water reactor (LWR) plants. However, based on literature review of existing research, radiation fluence limits of 1×10^{19} neutrons/cm^2 neutron radiation and 1×10^8 Gy (1×10^{10} rad) gamma dose are considered conservative radiation exposure levels beyond which concrete material properties may begin to degrade markedly (Ref. 17, 18, 19).

Further evaluation is recommended of a plant-specific program to manage aging effects of irradiation if the estimated (calculated) fluence levels or irradiation dose received by any portion of the concrete from neutron (fluence cutoff energy $E > 0.1$ MeV) or gamma radiation exceeds the respective threshold level during the subsequent period of extended operation or if plant-specific OE of concrete irradiation degradation exists that may impact intended functions. Higher fluence or dose levels may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and/or loss of mechanical properties of concrete from those fluence levels, at or above the operating temperature experienced by the concrete, and the effects are applied to the design calculations. Supporting calculations/analyses, test data, and other technical basis are provided to estimate and evaluate fluence levels and the plant-specific program. The acceptance criteria are described in BTP RLSB-1 (Appendix A.1 of this SRP-SLR).

Table 3.5.1 Item Number 3.5.1-097: The potential for reduction of strength, loss of mechanical properties, and cracking due to irradiation of reinforced concrete due to irradiation primarily concerns the reactor vessel shield wall, also called the sacrificial shield wall, around the reactor vessel in the drywell. The sacrificial shield wall is described in UFSAR Section C.4.6. The reactor vessel is supported from the bottom on a skirt on the reactor vessel support/pedestal structure, as described in UFSAR Section C.4.4, where the radiation exposure is much less than at the sacrificial shield wall along the reactor vessel belt line. The reactor vessel support/pedestal structure is a reinforced concrete hollow cylinder supporting both the reactor vessel and the sacrificial shield wall. The bottom and top portions of the shield wall is comprised of standard density concrete with limestone coarse

aggregate consisting of calcite and dolomite as well as quartz sand. The central portion of the sacrificial shield wall is comprised of high density, ilmenite concrete.

The estimated (calculated) fluence levels or irradiation dose received by any portion of the concrete from neutron (fluence cutoff energy E > 0.1 MeV) or gamma radiation does not exceed the respective threshold level limits of 1×10^{19} neutrons/cm^2 neutron radiation or 1×10^{10} rad gamma dose during the subsequent period of extended operation. Therefore, a plant-specific program to manage aging effects of irradiation is not required and the Structures Monitoring (B.2.1.34) program will be used to manage the potential for reduction of strength, loss of mechanical properties, and cracking due to irradiation of reinforced concrete due to irradiation near reactor vessel (sacrificial shield wall).

SLRA Table 4.2.~~2.~~1.1-1 shows a maximum fluence level of 2.23×10^{18} neutrons/cm^2 neutron radiation (fluence cutoff energy E > 1 MeV) at the inner diameter of the reactor vessel at the belt line for the 70 EFPY projected for the period of second period of extended operation. The maximum estimated fluence levels at the concrete are based upon determining the attenuation by the intervening reactor vessel shell and air gap and determining the neutron fluence levels at the energy levels of interest regarding potential concrete damage. The following is based upon EPRI report 3002008128, Revision 0, July 2016, "Structural Disposition of Neutron Radiation Exposure in BWR Vessel Support Pedestals".

- The following equation represents neutron attenuation through the thickness of the RPV:

  $f_{1T} = f_{0T}\,(e^{-0.33x})$ neutrons/cm^2 neutron radiation (fluence cutoff energy E > 1 MeV)

  Where:

  f_{1T} = neutron flux at the outside surface of the reactor vessel (1T) = 2.95×10^{17}
  f_{0T} = neutron flux at the inner surface of the reactor vessel (0T) = 2.23×10^{18}
  x = thickness of the reactor vessel = 6.125 inches

- The radiation exposure at the outside of the reactor vessel for neutrons with a fluence cutoff energy E > 0.1 MeV is estimated to be a factor of less than seven times the fluence values for neutrons with a fluence cutoff energy E > 1 MeV, which for PBAPS results in $2.95 \times 10^{17} \times 7 = 2.1 \times 10^{18}$ neutrons/cm^2 neutron radiation (fluence cutoff energy E > 0.1 MeV).

- The neutron exposure at the concrete of the reactor vessel shield wall is attenuated by the air gap between the reactor vessel and the reactor vessel shield wall and is conservatively reduced by 10 percent. As a result, the neutron exposure at the inside surface of the concrete shield wall, conservatively ignoring the steel on the inside of the shield wall, is = $0.9 \times 2.1 \times 10^{18} = 1.9 \times 10^{18}$ neutrons/cm^2 neutron radiation (fluence cutoff energy E > 0.1 MeV), which is less than the recommended radiation fluence limits of 1×10^{19} neutrons/cm^2 neutron radiation.

Two independent estimates of the gamma radiation levels were used to verify that recommended gamma radiation level would not be exceeded, which were based

on an EPRI report that used a NUREG/CR-5449 estimate of radiation levels at a BWR shield wall as well as a PBAPS calculation of gamma radiation streaming through sacrificial shield wall penetrations.

The following is based upon EPRI report 3002002676, revision 0, February 2014, "Expected Condition of Reactor Cavity Concrete after 80 Years of Radiation Exposure". In section 3.5 of the EPRI report, the gamma radiation dose on the inside of the shield wall is estimated for three plants, one of which is a BWR. The estimated gamma dose at the BWR plant for 80 years is 4.27×10^9 rad. The dose at 80 years was extrapolated from NUREG/CR-5449, June 1990, "Determination of the Neutron and Gamma Flux Distribution in the Pressure and Cavity of a Boiling Water Reactor". EPRI report 3002002676 has a comparison of various neutron fluence levels that can be used to assess radiation levels at different plants. The PBAPS neutron fluence levels are less than one half the neutron fluence levels for the plant evaluated in NUREG/CR-5449, which indicates that the gamma dose levels at the inside of the sacrificial shield wall at Peach Bottom are estimated to be less than the limit of 1×10^{10} rad.

A plant specific calculation was used where the gamma radiation dose rate at the inside of the sacrificial shield wall of 1.53×10^4 Rem/hour was calculated for use in determining the potential radiation dose through sacrificial shield wall penetrations. Considering Rem equivalent to rad and extrapolating for 40 EFPY at 100 percent capacity for the originally licensed power levels and 40 years at 100 percent capacity for the uprated power levels, and then considering that there will be only 70 EFPY at the end of the second period of extended operation, reveals that the gamma dose will be ***1×10^{10}*** ~~***9.7×10^9***~~ rad, which is ***at*** ~~***less than***~~ the recommended limit of 1×10^{10} rad, and conservatively ignores the shielding effect from the one quarter inch thick carbon steel liner on the inside of the shield wall.

Recent research on the gamma dose limit of 1×10^{10} rad reveals that this value may be overly conservative after subsequent reviews of previous test data. A recent paper published by I. Maruyama et al, Journal of Advanced Concrete Technology, Volume 15, 440-523 (2017), funded by the Japanese Regulator, concluded that there is no direct effect of gamma dose on concrete strength and recommends removing gamma dose limits. This paper concludes that previous studies that showed a decrease in concrete strength as a function of gamma dose were seeing an elevated temperature effect due to the high gamma flux in accelerated aging tests. Similar issues with the gamma dose limit of 1×10^{10} rad were also identified in NUREG/CR-7171, November 2013, "A Review of the Effects of Radiation on Microstructure and Properties of Concrete Used in Nuclear Power Plants".

SLR Reactor Vessel Support Steel Evaluation

In addition to the potential aging effects due to irradiation of reinforced concrete, a loss (or reduction) in fracture toughness due to irradiation embrittlement of the reactor vessel (RV) support steel is a potential aging effect considered in this subsection. NUREG-1509, May 1996, "Radiation Effects on Pressure Vessel Supports", is a resource for addressing the issue for SLR. Accordingly, a further evaluation of the RV supports for radiation induced embrittlement is provided below.

The reactor vessel is shown in UFSAR Figure 4.2.1. The RV support structures at PBAPS are described in UFSAR Section 4.2.4. The RV support structures consist of a cylindrical skirt attached to the bottom head of the reactor vessel and a lateral stabilizer at the top of the reactor vessel shield wall.

SLRA Figures 4.2.1-1 and 4.2.1-2 show the elevations of the reactor vessel belt line components. The radiation fields are significantly reduced above and below this region considering distance correction factors using the inverse square law as described in EPRI 3002008128, July 2016, "Structural Disposition of Neutron Radiation Exposure in BWR Vessel Support Pedestals". The RV support skirt is fabricated from plate steel conforming to ASME SA-302, Grade B. The UFSAR, Section 4.2.4.1 notes that the as-fabricated, initial NDT temperature is no higher than 40 °F. NUREG-1509, Section 4.2.1 notes that radiation embrittlement is not an issue for RV support skirts.

EPRI Report 3002014882, December 2018, "An Assessment of the Integrity of BWR Vessel Structural Steel Supports for Long-Term Operations", has recently been prepared to address irradiation of the RV support and applies to PBAPS, which is listed in Figure 3-1. The EPRI document evaluates the estimated maximum fluence levels and degree of embrittlement that was projected for the high stress (knuckle) region of the BWR reactor vessel supports. Also, the temperatures and loading conditions in the knuckle region were examined to determine whether irradiation induced embrittlement of the reactor vessel support steel could reduce the level of toughness and affect the margins against brittle fracture. The EPRI document concludes that the predicted level of embrittlement is minimal, using the appropriate embrittlement trend curve model for the BWR vessel supports after 80 years of plant operation. The predicted level of embrittlement is minimal since the fluence is low, the operating temperature is high, and the ductility of the skirt knuckle region is high. Therefore, the integrity of the reactor vessel supports is assured, and no additional aging management of reactor vessel supports beyond the current ASME Section XI, Subsection IWF (B.2.1.31) program is necessary for aging effects due to irradiation during the subsequent period of extended operation of PBAPS.

Accordingly, there is reasonable assurance that a loss (or reduction) of fracture toughness due to irradiation embrittlement will not affect the ability of the RV support steel to perform its component intended functions through the subsequent period of extended operation.

SLR Sacrificial Shield Wall- Structural Steel Evaluation

In addition to the potential aging effects due to irradiation of reinforced concrete, a loss (or reduction) in fracture toughness due to irradiation embrittlement of the nearby structural steel is a potential aging effect considered in this subsection. Specifically, the potential effects of irradiation on the steel elements of the sacrificial shield wall are addressed below in this further evaluation. The potential effects of irradiation on the concrete elements of the sacrificial shield wall and the reactor vessel supports are addressed in the previous subsections of this further evaluation.

As described in UFSAR Appendix C, Sections C.4.4 and C.4.6, the sacrificial shield wall is a 27-inch thick cylindrical structure that consists of twelve steel columns equally spaced and tied together by a 1/4-inch thick steel liner plates. Ring girders and transfer beams connect the columns together. Concrete was placed between the columns to provide radiation shielding. The sacrificial shield wall has an inside diameter of 24-feet and is approximately 49-feet high. For the seismic design, as shown in UFSAR Appendix C, Figure C.3.3, the sacrificial shield wall was modeled as a vertical beam anchored at the bottom to the top of the reactor pedestal with a lateral stabilizer at the top of the sacrificial shield wall. The lateral stabilizer consists of circular truss, which transfers loads through additional attachments to the reactor vessel, the top of the sacrificial shield wall, the interior and exterior of the drywell, and the interior of the shield wall outside of the drywell. The lateral stabilizer function is to transfer lateral forces applied at the top of the reactor vessel and the top of the sacrificial shield wall, through the drywell shell to the concrete shield wall outside of the drywell.

NUREG-1509 is a resource for addressing the potential effects of irradiation on the steel elements of the sacrificial shield wall for SLR. Accordingly, a further evaluation of the sacrificial shield wall for radiation induced embrittlement is provided below. Per NUREG-1509, the reduction in fracture toughness assessment of the sacrificial shield wall steel can be based on a transition temperature analysis, wherein a demonstration is made that there is adequate margin between the normal operating temperature and the ductile-to-brittle fracture mode transition temperature (commonly known as the NDT temperature for end-of-life (EOL) conditions. The transition temperature approach is based on the proposition that catastrophic failure by brittle fracture can be avoided by maintaining the normal operating sacrificial shield wall service temperature above the NDT temperature of the steel. When using the transition temperature to evaluate the sacrificial shield wall integrity, the NDT temperature at EOL should include the irradiation induced shift.

As described in the original construction specifications and confirmed in the material receipt records, the steel elements of the sacrificial shield wall, consisting of the columns, 1/4-inch thick steel liner plates, ring girders, and transfer beams, are fabricated from steel conforming to ASTM A36 low carbon steel. The assumed initial NDT temperature plus 1.3 σ, provided in

NUREG-1509 Table 4-1 and Table 4-2 for this material is 39°F. The original specification did not specify that any additional copper or nickel be incorporated into the ASTM A36 material and there are no chemical measurements for copper or nickel in material receipt records for the PBAPS shield walls made from ASTM A36 low carbon steel. The assumed chemical composition for the A36 steel shield wall (from NUREG/CR-6399, April 1977, "Results of Charpy V-Notch Impact Testing of Structural Steel Specimens Irradiated at ~30° C to 1 x 10^{16} neutrons/cm^2 in a Commercial Reactor Cavity", Table 2) is Cu = 0.05 weight %, Ni = 0.07 weight %, trace elements for A36 material. These are typical measured values from plate materials taken from the Trojan reactor and these chemistry values are representative of other shield wall plates.

NUREG-1509 provides a method for approximating the NDT shift by determining exposure in terms of displacements per atom (dpa), and then using Figure 3-1 of that reference to establish the irradiation induced shift of the NDT. By fitting the experimental data in NUREG-1509, a trend curve prediction model was developed for embrittlement shift versus displacements per atom (dpa) that incorporated the effects of flux and fluence, irradiation temperature, and gamma heating for application to the vessel supports as shown by the upper bound line in Figure 3-1 in NUREG-1509. That model included an upper bound transition temperature shift that was adjusted with zero-degree shift at a dpa of 10^{-5}. For the purpose of this evaluation of the sacrificial shield wall, use of the NUREG-1509 trend curve model for NDT shift versus dpa is very conservative since there is very little copper in the ASTM A36 materials and because the ratio of low energy neutrons to fast neutrons in the sacrificial shield wall is much smaller than that used in the test reactor experiments.

For the purposes of this further evaluation, fluence calculations were performed to confirm the attenuation effects through the RV internals, the RV and outward to the sacrificial shield wall. The peak fluence at the shield wall I.D. for 70 EFPY equates to a displacement per atom = 4.43 x 10^{-4} dpa.

The prediction of the potential irradiation induced NDT shift using the NUREG-1509 method shown in Figure 3-1 is described below. The potential irradiation induced NDT is a function of the dpa fluence as shown in the figure. The dashed upper bound curve is based on the fit to the experimental test data for reactor vessel carbon steel support materials (which did not include ASTM A36 materials) under low temperature, low flux neutron exposure conditions. For the sacrificial shield wall with a peak inner diameter surface fluence equating to 4.43 x 10^{-4} dpa, the corresponding NDT shift is 45°F. Therefore, the total adjusted NDT = Initial NDT + potential irradiation induced NDT shift = 39° + 45° = 84°F. This is well below the normal operating temperature in the drywell of 135°F.

Welding of the sacrificial shield wall was performed in accordance with the original design specification and the construction records that required the use of the SMAW welding process utilizing E-7018 and E-7028 electrodes, which conformed to specification AWS A5.1- 1964 and 1969 and do not incorporate a copper covering on the electrode. The original specification did not specifically add requirements for any additional copper or nickel and

there are no chemical measurements for copper or nickel in material receipt records for the weld rods used for the PBAPS sacrificial shield walls beyond the standard material requirements, which do not include requirements for copper or nickel. As a result, the weld materials are similar to the ASTM A36 materials for the purposes of this further evaluation, and the same conclusions are made regarding the potential effects of irradiation induced embrittlement for the weld materials incorporated into the sacrificial shield wall as were made regarding the sacrificial shield wall steel elements.

Therefore, an evaluation of the potential effects of irradiation on the steel elements of the sacrificial shield wall was performed for PBAPS, using the projected fluence values for 70 EFPY and an adjusted NDT for the steel materials using the methodology in NUREG-1509. The evaluation demonstrates that there is adequate margin between the normal operating temperature and the ductile-to-brittle fracture mode transition temperature that was adjusted for the potential effects due to irradiation. The evaluation concludes that the potential effects of irradiation on the steel elements of the sacrificial shield wall materials, including the welding material, are not significant. As a result, the integrity of the sacrificial shield wall is assured, and no additional aging management of the sacrificial shield wall beyond the current Structures Monitoring (B.2.1.34) program is necessary for aging effects due to irradiation during the subsequent period of extended operation of PBAPS.

Accordingly, there is reasonable assurance that a loss (or reduction) of fracture toughness due to irradiation embrittlement will not affect the ability of the sacrificial shield wall support steel to perform its component intended functions through the subsequent period of extended operation. Since the radiation dose to the sacrificial shield wall bounds other Class 1 steel structures at PBAPS, and the aging effects due to irradiation are not significant for the sacrificial shield wall, a plant specific program is not necessary in order to manage the aging effects due to irradiation of structural steel components.