APPENDIX Q LICENSE RENEWAL UFSAR SUPPLEMENT

Q.0 Introduction

This appendix contains the UFSAR Supplement as required by 10 CFR 54.21(d) for the Peach Bottom License Renewal Application (LRA). The NRC issued SER NUREG-1769 that provides their safety evaluation of the Peach Bottom LRA.

The aging management activity descriptions presented in this appendix represent commitments for managing aging of the in-scope systems, structures and components during the period of extended operation.

As part of the license renewal effort, it had to be demonstrated that the aging effects applicable for the components and structures within the scope of license renewal would be adequately managed during the period of extended operation.

In many cases, existing activities were found adequate for managing aging effects during the period of extended operation. In some cases, aging management reviews revealed that existing activities required enhancement to adequately manage applicable aging effects. In a few cases, new activities were developed to provide added assurance that aging effects are adequately managed.

Plant operating at extended power uprate (EPU) for the period of extended operation has been incorporated in this appendix.

Plant operation under MELLLA+ core flow regime has been incorporated in this appendix.

<u>Activities Credited for Managing Aging in the Renewal Term</u>

PBAPS has numerous activities that detect and monitor aging effects. This supplement to the UFSAR only describes those activities, which PBAPS is crediting for the purposes of complying with the license renewal rule.

Each aging management activity presented in this Appendix is characterized as one of the following:

- Existing Activity: An activity in existence prior to license renewal approval that will continue to be implemented during the period of extended operation. (Although some activities were existing at the time of LRA issue, these had to be modified due to Requests for Additional Information (RAIs) and Safety Evaluation Report (SER) open item responses. To maintain the same numbering scheme, these are still considered existing activities rather than enhanced activity.)

- Enhanced Activity: An activity in existence prior to license renewal approval that will be modified during the extended period of operation. Enhancements were implemented as discussed in this Appendix.

- New Activity: An activity that did not exist prior to license renewal approval, which will manage aging during the extended period of operation. These activities were implemented as described in this Appendix.

- Time Limited Aging Analyses Activity: An activity that has been credited by a time-limited aging analysis as described in this Appendix.

The commitment tracking number for each aging management activity is identified in parenthesis against each activity.

See Section Q.7 for newly identified items under 10 CFR 54.37(b).

Q.1 EXISTING AGING MANAGEMENT ACTIVITIES

Q.1.1 Flow Accelerated Corrosion Program (2701330-60 / T04338)

The PBAPS Flow Accelerated Corrosion (FAC) Program activities manage loss of material in pipes and fittings by monitoring the condition of piping susceptible to FAC induced wall thinning. The FAC Program provides for prediction of the amount of wall thinning in carbon steel pipes and fittings through analytical evaluations and periodic examinations of locations most susceptible to FAC induced loss of material. The program includes analyses to determine critical locations, baseline inspections to determine the extent of thinning at these locations, and follow-up inspections to confirm the predictions. The FAC Program provides reasonable assurance that loss of material of carbon steel pipe and fittings is detected and addressed prior to loss of intended function of the piping.

Q.1.2 Reactor Coolant System Chemistry (2701330-58 / T04336)

PBAPS Reactor Coolant System (RCS) chemistry activities manage loss of material and cracking of components exposed to reactor coolant and steam through measures based on BWRVIP-190, "BWR Water Chemistry Guidelines," that monitor and control reactor coolant chemistry. These activities include monitoring and controlling of Reactor Coolant water chemistry to ensure that known detrimental contaminants are maintained within pre-established limits. Reactor Coolant is monitored for indications of abnormal chemistry conditions. If such indications are found, then measurements of impurities are conducted to determine the cause, and actions are taken to address the abnormal chemistry condition. Whenever corrective actions are taken to address an abnormal chemistry condition, sampling is utilized to verify the effectiveness of these actions. The RCS chemistry activities provide reasonable assurance that intended functions of components exposed to reactor coolant and steam are not lost due to loss of material or cracking aging effects.

Q.1.3 Closed Cooling Water Chemistry (2701330-58 / T04336)

The PBAPS Closed Cooling Water (CCW) chemistry activities manage loss of material, cracking and reduction of heat transfer in components exposed to closed cooling water through measures that monitor and control cooling water chemistry. These activities include periodic monitoring and controlling of chemistry parameters and corrosion inhibitors. If parameter limits are exceeded, corrective actions are taken to restore parameters within the acceptable range. The CCW chemistry activities provide reasonable assurance that the intended functions of components in a CCW environment are not lost due to loss of material, cracking, or reduction of heat transfer aging effects.

Q.1.4 Demineralized Water and Condensate/Refueling Water Storage Tank Chemistry Activities (2701330-58 / T04336)

PBAPS Demineralized Water and Condensate Storage/Refueling Water Tank chemistry activities manage loss of material and cracking of components exposed to demineralized water, Condensate Storage Tank (CST) water, and Refueling Water Storage Tank (RWST) water in the RCIC, HPCI, CRD, Core Spray, Standby Liquid Control, Demineralized Water, RHR, Refueling Water Storage and Transfer, PASS, and Condensate Storage Systems. In addition, CST chemistry activities manage reduction in heat transfer in the HPCI gland seal condenser, and the RCIC and HPCI turbine lubricating oil coolers. The demineralized water, CST water, and RWST water are monitored periodically to assure that purity is maintained within pre-established limits. If parameter limits are exceeded, corrective actions are taken to restore parameters within the acceptable range. These chemistry activities provide reasonable assurance that intended functions of in-scope components exposed to demineralized water, CST water, and RWST water are not lost due to loss of material, cracking, or reduction of heat transfer aging effects.

Q.1.5 Torus Water Chemistry Activities (2701330-58 / T04336)

PBAPS Torus water chemistry activities manage loss of material and cracking of components exposed to Torus grade water in the RHR, HPCI, RCIC, Core Spray and Main Steam systems. In addition, Torus water chemistry activities manage cracking of stainless steel component supports submerged in Torus grade water, and reduction of heat transfer in RHR heat exchangers. Water from the Torus is monitored periodically to assure that purity is maintained within pre-established limits. Torus water chemistry activities provide reasonable assurance that intended functions of components exposed to Torus grade water are not lost due to loss of material, cracking, or reduction of heat transfer aging effects.

Q.1.6 Fuel Pool Chemistry Activities (2701330-58 / T04336)

PBAPS Fuel Pool chemistry activities manage loss of material for fuel Pool gates, fuel storage racks, Fuel Pool liner, component supports, fuel preparation machines, Refueling Platform mast, and loss of material and cracking for Fuel Pool Cooling and Cleanup System components exposed to Fuel Pool water. Fuel Pool water is monitored periodically to assure that purity is maintained within pre-established limits. Fuel Pool chemistry activities provide reasonable assurance that intended functions of components contacted by Fuel Pool water are not lost due to loss of material or cracking aging effects.

Q.1.7 High Pressure Service Water Radioactivity Monitoring Activities (2701330-63 / T04341)

PBAPS High Pressure Service Water (HPSW) radioactivity monitoring activities manage loss of material and cracking in RHR heat exchangers through routine sampling and isotopic analysis of the HPSW system water contained within the RHR heat exchangers to confirm the absence of radioactive isotopes that do not occur naturally. HPSW radioactivity monitoring activities provide reasonable assurance that loss of material and cracking are detected and addressed prior to loss of intended function.

Q.1.8 Inservice Inspection (ISI) Program (2701330-65 / T04343)

The Inservice Inspection (ISI) aging management program, as augmented to address the requirements of GL 88-01, consists of those portions of the PBAPS ISI program that are being utilized for managing aging in pressure retaining piping and components in the scope of license renewal. However, the reactor pressure vessel components and internals in the PBAPS ISI program are not included in the ISI aging management program. PBAPS complies with the requirements of ASME Section XI Code Edition and Addenda per the ISI program, and includes requirements for inspections of ASME XI Class 1, 2, and 3 (Per Reg. Guide 1.26 guidance) pressure retaining components. Age related degradation identified during inspections of Class 1 portions of the Reactor Recirculation System will be evaluated for applicability to the non-safety related portions of the Reactor Recirculation System that is included in the scope of license renewal. In addition, it provides for condition monitoring of ASME XI Class 1,2 and 3 piping and equipment supports and integral support anchors. The ISI program provides reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

Q.1.9 Primary Containment Inservice Inspection Program (2701330-67 / T04345)

The Primary Containment ISI program consists of inspections that manage loss of material in the primary containment for Class MC pressure-retaining components, their integral attachments, and Class MC component supports; and loss of sealing for the Drywell internal moisture barrier at the juncture of the Containment wall and the concrete floor. PBAPS complies with ASME Section XI Code, Subsection IWE, Edition and Addenda per the ISI program. The Primary Containment ISI program provides reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

Q.1.10 Primary Containment Leakage Rate Testing Program (2701330-48 / T04325)

The Primary Containment Leakage Rate Testing Program is that portion of the PBAPS Primary Containment Leakage Rate Testing Program that is being credited for license renewal. The Primary Containment Leakage Rate Testing Program provides for aging management of pressure boundary degradation due to loss of material in a wetted gas environment in containment atmosphere control and dilution, RHR, and Primary Containment Isolation Systems penetrating Primary Containment. The Primary Containment Leakage Rate Testing Program also manages change in material properties and cracking of gaskets and O-rings for the Primary Containment pressure boundary access penetrations. The program complies with the requirements of 10CFR50 Appendix J, Option B, and provides reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

Q.1.11 Inservice Testing (IST) Program (2701330-57 / T04335)

The Inservice Testing (IST) aging management program is that portion of the PBAPS IST program that is being credited for license renewal. The IST aging management program manages flow blockage of system components from the ECW pump through the ESW and ECW system piping to the ECT. In addition, the program manages reduction of heat transfer of the RHR heat exchangers through flow testing of the torus water path. IST program activities are conducted in accordance with the ASME O&M Code. IST program activities provide reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

Q.1.12 Reactor Materials Surveillance Program (2701330-68 / T04346)

The PBAPS Reactor Materials Surveillance (RMS) program manages loss of fracture toughness in the Reactor Pressure Vessel beltline region consistent with the requirements of 10 CFR 50, Appendix H and ASTM E185. Compliance with 10CFR50, Appendix H is demonstrated through the integrated surveillance program (ISP) that meets the technical requirements documented within the latest NRC-approved version of BWRVIP-86. The RMS program provides for periodic withdrawal and testing of in-vessel capsules to monitor the effects of neutron embrittlement on the Reactor Vessel beltline materials. The results of this testing are used to determine plant operating limits. The RMS program contains sufficient dosimetry and materials to monitor irradiation embrittlement during the period of extended operation and provides reasonable assurance that aging effects are detected and addressed prior to loss of intended function.

Q.1.13 CORRECTIVE ACTION PROGRAM (2701330-79 / T04473)

The Corrective Action Program provides for evaluation of aging effects and significant operating events and requires that reasonable actions be taken to enhance programs and activities to prevent future occurrences. The plant condition reporting process applies to all plant structures, systems and components within the scope of license renewal. Corrective action is initiated following the identification of conditions adverse to quality. An effectiveness review is completed for all root cause analysis corrective actions to prevent recurrence and other items as assigned by the PBAPS Management Review Committee. If corrective actions to prevent recurrence are determined to be ineffective, this deficiency is addressed by the existing condition report or a new condition report is originated to address the deficiency and initiate resolution. Administrative controls are in place for existing new and enhanced aging management programs and activities. As a minimum, these programs and activities are performed in accordance with written procedures. Those procedures are reviewed and approved in accordance with PBAPS’s 10CFR50, Appendix B, QA Program.

Q.1.14 Crane Inspection Activities (2701330-51 / T04328)

PBAPS crane inspection activities manage loss of material for the structural members, rails, and rail anchorage for the

Circulating Water Pump Structure gantry crane, and rails and monorails for the cranes and hoists located in a sheltered environment. These crane inspections provide reasonable assurance that loss of material is detected and addressed prior to loss of intended function.

Q.1.15 Conowingo Hydroelectric Plant (Dam) Aging Management Program (2701330-85 / T04504)

The Conowingo Hydroelectric Plant dam is subject to the FERC 5-year inspection program. This program consists of a visual inspection by a qualified independent consultant approved by Federal Energy Regulatory Committee (FERC), and is in compliance with Title 18 of the Code of Federal Regulations, Conservation of Power and Water Resources, Part 12 (Safety of Water Power Projects and Project Works), Subpart D (Inspection by Independent Consultant). The NRC has found that mandated FERC 5-year inspection programs are acceptable for aging management.

Q.1.16 Maintenance Rule Structural Monitoring Program (2701330-66 / T04344)

The Maintenance Rule Structural Monitoring program is that portion of the PBAPS Structural Monitoring Program that is being credited for license renewal. The Maintenance Rule Structural Monitoring Program complies with 10CFR50.65 and utilizes visual inspections in managing aging effects for concrete and grout in accessible areas, masonry block walls, carbon steel structures and components, and hazard barrier seals within the scope of license renewal that are not covered by other existing inspection programs. Concrete and masonry block walls are monitored for loss of material, cracking, and change in material properties. Grout is monitored for cracking and carbon steel structures and components are monitored for loss of material, due to corrosion. Hazard barrier penetration seals and expansion joints are monitored for change in material properties, delamination and separation, and cracking. The Maintenance Rule Structural Monitoring Program was enhanced to include inspection of carbon steel component supports (other than ASME XI Class 1, 2, 3, and ASME XI Class MC component supports) and SBO structural components for loss of material. Maintenance Rule Structural Monitoring Program activities provide reasonable assurance that aging effects are detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.1.17 Electrical Cables Not Subject to 10CFR50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (2701330-82 / T04476)

This aging management activity applies to electrical cables used in the Local Power Range Monitor, and Wide Range Neutron Monitor instrumentation circuits. The periodic review of calibration test results is used to identify the potential existence of aging degradation. When an instrumentation circuit is found to be significantly out of calibration, additional evaluation is performed on the circuit, including the cable, as required. This activity provides reasonable assurance that the intended functions of electrical cables that are not subject to the environmental qualification requirements of 10CFR50.49 and are used in instrumentation circuits with sensitive, low-level signals exposed to adverse localized environments caused by heat, radiation or moisture will be maintained consistent with the current licensing basis through the period of extended operation.

Q.2 ENHANCED AGING MANAGEMENT ACTIVITIES

Q.2.1 Lubricating and Fuel Oil Quality Testing Activities (2701330-49 / T04326, 2701330-50 / T04327)

Lubricating and Fuel Oil quality testing activities manage loss of material, cracking and reduction of heat transfer in components that contain or are exposed to lubricating oil or fuel oil. Lubricating and Fuel Oil quality testing activities provide for sampling and testing of lubricating oil in components in Emergency Diesel Generator (EDG), High Pressure Coolant Injection (HPCI), High Pressure Service Water (HPSW), Core Spray (CS), and Reactor Core Isolation Cooling (RCIC) systems. Lubricating and Fuel Oil quality testing activities also provide for sampling and testing of fuel oil in the EDG and Diesel Driven Fire Pump fuel oil systems. Lubricating and Fuel Oil quality testing activities include sampling and analysis of lubricating oil and fuel oil for detrimental contaminants. The Diesel Driven Fire Pump fuel oil sampling methods were enhanced to improve water detection capabilities. Analyses of the Diesel Driven Fire Pump and EDG fuel oil samples were enhanced to add testing for microbes in any water detected. The Lubricating and Fuel Oil quality testing activities provide reasonable assurance that aging effects on system components will be detected and addressed prior to loss of intended function of the components. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.2.2 Boraflex Management Activities (2701330-53 / T04330)

Since NETCO SNAP-IN® rack inserts have been fully installed in both Peach Bottom Unit 2 and Unit 3 Spent Fuel Pool racks, Boraflex is no longer credited as a neutron absorbing material.

Q.2.3 Ventilation System Inspection and Testing Activities (2701330-47 / T04324)

PBAPS Ventilation System inspection and testing activities manage aging of filter plenum access door seals and fan flex connections in the Standby Gas Treatment System and the Control Room Ventilation System. These activities also include inspections of fan flex connections for the Standby Gas Treatment System, the Control Room Ventilation System, the Battery Room and Emergency Switchgear Ventilation System exhaust fans, and the ESW booster pump room ventilation supply fans. These activities were enhanced to add inspections of fan flex connections in the Diesel Generator building ventilation system, the ESW/HPSW Compartment Ventilation System and the Battery Room and Emergency Switchgear Ventilation system supply fans. PBAPS ventilation system inspection and testing activities provide reasonable assurance that change in material properties will be detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.2.4 Emergency Diesel Generator Inspection Activities (2701330-55 / T04332)

The Emergency Diesel Generator (EDG) inspection activities provide for condition monitoring of EDG equipment within the scope of license renewal that are exposed to a gaseous, closed cooling water, lubricating oil or fuel oil environment. Loss of material in the starting air system air receivers is mitigated by daily removal of any accumulation of condensate. Loss of material and cracking in lubricating oil and fuel oil systems is mitigated by periodic inspections performed for underground storage tanks. Visual inspections for change in material properties of flexible hoses in the starting air system and the cooling water system are performed in accordance with a PBAPS procedure in connection with periodic EDG maintenance. This procedure was enhanced to require inspections of the lubricating oil system and fuel oil system flexible hoses for a change in material properties. Loss of material in the EDG exhaust silencer is managed by periodic disassembly, cleaning, and inspection of an automatic drain trap to ensure its functionality in preventing condensation build up. The EDG inspection activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function of the components. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.2.5 Outdoor, Buried and Submerged Component Inspection Activities (2701330-52 / T04329)

The outdoor, buried, and submerged component inspection activities provide for loss of material and cracking aging management of external surfaces of components subject to outdoor, buried, and raw water external environments. (Separately, the ISI program provides for monitoring of pressure boundary integrity for outdoor and buried components through pressure tests, flow tests, and inspections.) The submerged components include HPSW, ESW, ECW, and Fire Protection System pumps. HPSW and ESW system manual discharge pond isolation valves, Condensate Storage System and Refueling Water Storage and Transfer System piping and valves, the external surfaces of the Condensate Storage Tanks (CSTs), the external surfaces of the Refueling Water Storage Tank (RWST), and the piping insulation jacketing at the CSTs and RWST are the components exposed to the outdoor environment. The buried components include HPSW, ESW, ECW, fire protection, and EDG fuel oil system piping, Fire Protection System fire main isolation valves, the EDG fuel oil storage tanks, the SGTS exhaust to the main stack, Condensate Storage System piping and valves, and the undersides of the CSTs and RWST which are in direct contact with compacted fill. The outdoor, buried, and submerged component inspection activities are implemented in accordance with PBAPS maintenance procedures and routine test procedures. The scope of components covered by these activities were enhanced to include periodic visual inspection of the external surfaces of the CSTs and RWST, periodic visual inspection of the ECW pump casing and casing bolts, visual inspection of buried commodities whenever they are uncovered during excavation, enhanced inspections of the RWST and CST tank bottoms (as a result of plant and industry operating experience), and inspection of cast iron sluice gates in raw water environment. These inspection activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

<u>Q.2.6 Door Inspection Activities</u> (2701330-54 / T04331)

The Door Inspection activities provide for managing the aging effects for hazard barrier doors that are exposed to the outdoor and sheltered environments. The aging management review determined that the activities were enhanced to include additional doors exposed to sheltered environment. In addition, the activities were enhanced to include inspection for loss of material, due to corrosion, in hazard barrier doors. The door inspection activities also provide for managing the aging effects for gaskets associated with water-tight hazard barrier doors in both outdoor and sheltered environments. The inspection activities consist of condition monitoring of the gaskets associated with water-tight hazard barrier doors on a periodic basis. The Hazard Barrier Doors Inspection activities are condition monitoring activities that utilize inspections to provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.2.7 Reactor Pressure Vessel and Internals ISI Program
(2701330-69 / T04347)

The BWR Vessels and Internals Project (BWRVIP) guidelines are implemented through the Reactor Pressure Vessel and Internals ISI program. The Reactor Pressure Vessel and Internals ISI program is that part of the PBAPS ISI program that provides for condition monitoring of the Reactor Vessel and Internals using guidance provided by the BWRVIP and the BWR Owners Group alternate BWR Feedwater nozzle inspection requirements. The PBAPS ISI program complies with requirements of an NRC approved Edition of the ASME Section XI Code, or approved alternative, and is implemented through a PBAPS specification. The PBAPS ISI program has been augmented to include various additional requirements, including those from the BWRVIP guidelines and the BWR Owners Group (BWROG) alternative to NUREG-0619 augmented inspection of Feedwater nozzles for GL 81-11 thermal cycle cracking. The Reactor Pressure Vessel and Internals ISI program were enhanced to assure that inspections are consistent with the relevant BWRVIP program criteria and NRC safety evaluation reports. The Reactor Pressure Vessel and Internals ISI program were enhanced to require inspection of top guide similar to the inspection of CRDH guide tubes. The Reactor Pressure Vessel and Internals ISI Program utilizes the staff-approved BWRVIP core shroud inspection and evaluation guidelines program. The program utilizes early detection, evaluation and corrective actions that provide reasonable assurance that aging effects of Reactor Vessel components and internals will be detected and addressed prior to loss of intended function. Program enhancements are implemented as the BWRVIP guidelines are revised. Top guide inspection enhancement were implemented prior to the end of the initial operating license term for PBAPS.

Reactor Pressure Vessel And Internals BWRVIP Document Applicability

Reactor Pressure Vessel Components	Reference
Reactor pressure vessel components	BWRVIP-74
Vessel shells	BWRVIP-05
Shroud support attachments	BWRVIP-38
Nozzle safe ends	BWRVIP-74
Core support plate	BWRVIP-25
Core ΔP / SLC nozzle	BWRVIP-27
Core spray attachments	BWRVIP-48
Jet pump riser brace attachments	BWRVIP-48
Other attachments	BWRVIP-48
CRDH stub tubes	BWRVIP-47
ICM Housing penetrations	BWRVIP-47
Instrument penetrations	BWRVIP-49
Reactor Internals Components	
Shroud support	BWRVIP-38
Shroud	BWRVIP-76
Core support plate	BWRVIP-25
Core ΔP / SLC line	BWRVIP-27
Access hole covers	(Note 1)
Top guide (Note 2)	BWRVIP-26
Core spray lines	BWRVIP-18
Core spray spargers	BWRVIP-18
Jet pump assembly	BWRVIP-41
CRDH stub tubes	BWRVIP-47
CRDH guide tubes	BWRVIP-47
In-core housing guide tubes, LPRM & WRNMS dry tubes	BWRVIP-47
Note 1. GE SIL 462 for Unit 2 only.	
Note 2: The Reactor Pressure Vessel and Internals ISI program were enhanced to require inspection of top guide similar to the inspection of CRDH guide tubes.	

<u>Q.2.8 GL 89-13 Activities</u> (2701330-56 / T04333)

The GL 89-13 activities provide for management of loss of material, cracking, flow blockage, and reduction of heat transfer aging effects in cooling water piping and components that are tested and inspected in accordance with the guidelines of NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety Related Equipment”. The GL 89-13 activities include both condition monitoring and mitigating activities for managing aging effects in HPSW, ESW, and the ECW systems and in other systems' components using raw water as a cooling medium. System and component testing, visual inspections, UT, and biocide treatments are conducted to ensure that aging effects are managed such that system and component intended functions are maintained. Maintenance procedures were enhanced to require inspection for specific signs of degradation, including corrosion, excessive wear, cracks and Asiatic clams. Also additional piping locations were added to the UT inspection program. The GL 89-13 activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.2.9 Fire Protection Activities (2701330-64 / T04342)

The Fire Protection activities provide for inspections, monitoring, and performance testing of fire protection systems and components to detect aging effects prior to loss of intended function. Degradation of Fire Protection systems and components due to corrosion buildup, biofouling, and silting are detected by performance testing based on NFPA 24 standards. Periodic and maintenance inspections detect corrosion, fouling, and cracking in system components due to internal and external environment aging effects and detect aging effects in fire barriers. Monitoring of system pressure detects system leakage due to both internal and external aging effects. The scope of fire protection activities were enhanced. The activities require additional inspection requirements for deluge valves in the power block sprinkler systems[1], testing of sprinklers that have been in service for 50 years, inspection of Diesel Driven Fire Pump exhaust systems, inspection of Diesel Driven Fire Pump fuel oil system flexible hoses, inspection of fire doors for loss of material, perform a one-time test of a cast iron Fire Protection component for loss of material due to selective leaching and inspection of conduits in an outdoor environment that support cables credited for fire safe shutdown.

The Fire Protection activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

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[1] Inspections will be performed through deluge valve reset procedures or 5-year PM activities as applicable based on the valve style installed.

Q.2.10 HPCI and RCIC Turbine Inspection Activities (2701330-62 / T04340)

The HPCI and RCIC turbine inspection activities provide for aging management of the HPCI and RCIC turbine casings that are exposed to a wetted gas environment. The HPCI turbine inspection activities additionally provide for condition monitoring of components exposed to a lubricating oil environment. The inspection activities perform assessments of components for loss of material aging effects. The HPCI and the RCIC turbine inspection activities are performed periodically in connection with turbine maintenance. The HPCI and RCIC turbine inspection activities provide reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.2.11 Susquehanna Substation Wooden Pole Inspection Activity (2701330-70 / T04348)

The Susquehanna Substation Wooden Pole Inspection activity manages the aging effects of loss of material and change in material properties for a wooden takeoff pole at the Susquehanna Substation. This pole provides the structural support for the conductors connecting the substation to the submarine cable that is used to transmit the alternate AC power for PBAPS from the Conowingo Hydroelectric Plant in compliance with the requirements of 10 CFR 50.63 for coping with Station Blackout. The inspection activity were enhanced to ensure that it is performed every ten years in accordance with corporate specification. The wooden pole inspection activity provides reasonable assurance that aging effects will be detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.2.12 Heat Exchanger Inspection Activities (2701330-61 / T04339)

The heat exchanger inspection activities provide for periodic component visual inspections and cleaning of heat exchangers and coolers that are outside the scope of NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety Related Equipment”. These activities manage aging effects of loss of material, cracking, and reduction of heat transfer effects for the HPCI gland seal condenser, the HPCI turbine lube oil cooler, and the RCIC turbine lube oil cooler. These activities were enhanced to require periodic inspection of the HPCI gland seal condenser tube side internals. These inspections provide reasonable assurance that aging effects are detected and addressed prior to loss of intended function. Activity enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.3 NEW AGING MANAGEMENT ACTIVITIES

Q.3.1 Torus Piping Inspection Activities (2701330-71 / T04349)

The PBAPS Torus piping inspection activities will provide for identification of loss of material in carbon steel piping located at the water-gas interface in the Torus compartment of the primary containment by monitoring the condition of a representative sample of the piping at a susceptible location. These activities included a one-time inspection of the wall thickness of selected torus piping. The scope and frequency of subsequent examinations will be based on the results of the initial inspection sample. Torus piping inspection activities provide reasonable assurance that loss of material will be detected and addressed prior to loss of intended function. Torus piping inspection activities were implemented prior to the end of the initial operating license term for PBAPS.

Q.3.2 FSSD Cable Inspection Activity (2701330-59 / T04337)

PBAPS Fire Safe Shutdown (FSSD) cable inspection activities will manage change in material properties of the PVC-insulated FSSD cables located in the Drywell by monitoring the condition of a representative sample of the cables. FSSD cable inspection activities will identify anomalies in the PVC insulation surface that are precursor indications of loss of material properties of the PVC insulation. These activities provide reasonable assurance that loss of material properties of the PVC-insulated cables will be detected and addressed prior to loss of their intended function. The FSSD cable inspection activities were implemented prior to the end of the initial operating license term for PBAPS.

Q.3.3 Non-EQ Accessible Cable Aging Management Activity (2701330-81 / T04475)

The Non-EQ accessible cable aging management activity visually inspects all cables and connections in accessible areas (easily approached and viewed) in the potential adverse localized environment. The Non-EQ accessible cable aging management activity is performed once every ten years, beginning prior to the period of extended operation. This inspection activity provides reasonable assurance that the intended functions of electrical cables and connections that are not subject to the environmental qualification requirements of 10CFR50.49 and are exposed to adverse localized environments caused by heat or radiation will be maintained consistent with the current licensing basis through the period of extended operation.

Q.3.4 One-Time Piping Inspection Activities (2701330-80 / T04474)

The PBAPS one-time piping inspection activities provide for identification of loss of material or cracking, as applicable for the system material and environment, by monitoring the condition of a representative sample of the piping at a susceptible location. The inspection activities confirmed the pressure integrity of the following piping systems:

- Standby Liquid Control (suction side of pumps between the pumps and the solution tank).
- Auxiliary Steam (piping in Units 2 & 3 RB, Units 2 & 3 Reactor Auxiliary Bay (RAB), DG Bldg., R/W Bldg., Nitrogen Storage Bldg., Circ Water Pumphouse)
- Plant Equipment and Floor Drains (piping in Units 2 & 3 RB, Units 2 & 3 Drywell)
- Service Water (piping in Units 2 & 3 RB, Units 2 & 3 RAB)
- Radiation Monitoring (piping in Units 2 & 3 RB, Units 2 & 3 RAB, and DG Building)
- Reactor Recirculation (RPV bottom head drain piping)
- RPV Instrumentation (Reactor head vent line to Wide Range Level Instrumentation condensing chamber)
- Fuel Pool Cooling (piping in Units 2 & 3 RB)

The one-time piping inspection activities were implemented prior to the end of the initial operating license term for PBAPS Unit 2 and Unit 3.

Q.3.5 INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS (2701330-83 / T04477)

The non-EQ inaccessible cable aging management activity applies to medium-voltage cables within the scope of license renewal that are exposed to significant moisture simultaneously with significant voltage. A representative sample of in-scope, medium-voltage cables is tested to provide an indication of the condition of the conductor insulation. The specific type of test performed was determined prior to the initial test. This activity will provide reasonable assurance that aging effects are detected and addressed such that the intended function will be maintained for the period of extended operation. This activity was implemented prior to the end of the initial operating license term for PBAPS.

Q.3.6 FUSE HOLDER AGING MANAGEMENT ACTIVITY (2701330-84 / T04478)

DELETED – SECOND LICENSE RENEWAL DETERMINED THIS ACTIVITY IS NOT NEEDED.

Appendix Q.3.7 Selective Leaching

The Selective Leaching aging management activity is a new condition monitoring activity for SSCs within the scope of License Renewal and subject to Aging Management Review that may be susceptible to loss of material due to selective leaching. This activity is being put into place due to internal Operating Experience that revealed the presence of Selective Leaching in certain SSCs. The PBAPS Selective Leaching aging management activity is only applied to those SSCs with the material and environment combinations where Selective Leaching has been found to exist at PBAPS. Susceptible materials are gray cast iron in raw water and soil environments only. A new procedure has been developed and existing procedures have been revised for the Selective Leaching aging management activity to perform opportunistic visual inspections supplemented by mechanical examination techniques {such as chipping and scraping} followed by confirmatory hardness measurements on SSCs susceptible to selective leaching.

Q.4 TIME-LIMITED AGING ANALYSES ACTIVITIES

Q.4.1 Environmental Qualification Activities (2701330-74 / T04412)

PBAPS Environmental Qualification (EQ) program ensures maintenance of qualified life for the electrical equipment important to safety within the scope of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." An aging limit (qualified life) is established for equipment within the scope of the EQ program and an appropriate action such as replacement or refurbishment is taken prior to or at the end of the equipment qualified life so that the aging limit is not exceeded. The PBAPS EQ program activities establish, demonstrate and document the level of qualification, qualified configuration, maintenance, surveillance and replacement requirements necessary to apply the qualification conclusions and the equipment qualified life. As permitted by License Amendment 321/324, dated 10/25/2018, to implement 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors", the station may voluntarily comply with the requirements of 10 CFR 50.69 as an alternative to compliance with 10 CFR 50.49 requirements for components classified as RISC-3 and RISC-4.

Q.4.2 Fatigue Management Activities (2701330-73 / T04411)

The Fatigue Management Program counts fatigue stress cycles and tracks fatigue usage factors. The program was enhanced to broaden its scope and update implementation methods, and will consist of analytical methods to determine stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles, and automated calculation and tracking of fatigue cumulative usage factors (CUFs). The program will calculates and track CUFs for bounding locations in the reactor pressure vessel (RPV), RPV internals, Group I piping, and containment torus. The Fatigue Management Program enhancements were implemented prior to the end of the initial operating license term for PBAPS.

Q.5 TIME-LIMITED AGING ANALYSES SUMMARIES

In the descriptions of this section, Groups I, II, and III are the PBAPS pressure boundary safety groups described in UFSAR Appendix A, Section A.2

Q.5.1 Reactor Vessel Neutron Embrittlement

The PBAPS Units 2 and 3 Reactor Vessels are described in UFSAR Sections 3.3 and 4.2. Reactor Vessel materials are subject to embrittlement, primarily due to exposure to neutron radiation. Reactor Vessel neutron embrittlement is a TLAA.

Q.5.1.1 Reactor Vessel Neutron Embrittlement

The Reactor Vessel materials are subject to embrittlement, primarily due to exposure from neutron radiation. Calculations for end-of-life fluence for a 60-year licensed operating period (54 EFPYs) using the GE fluence calculation methodology (NEDC-32983P-A, Revision 2, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations") were performed. NEDC-32983P-A was approved by the NRC in a letter dated November 17, 2005, from H. N. Berkow (NRC) to G. B. Stramback (GE). For Units 2 and 3, the 54 EFPY RPV peak fluence predictions were as follows: Unit 2 – 1.62 x 10^{18} n/cm^2 at the inner vessel wall and 1.12 x 10^{18} n/cm^2 at 1/4 T locations. Unit 3 – 1.54 x 10^{18} n/cm^2 at the inner vessel wall and 1.07 x 10^{18} n/cm^2 at 1/4 T location. Analyses have been performed that use these fluence results to address the following:

- Upper Shelf Energy
- P-T Limit Curves
- Reactor Vessel Circumferential Weld Examination Relief
- Reactor Vessel Axial Weld Failure Probability

The Technical Requirements Manual (TRM) contains current P-T curves. Reference 6 contains the current fracture toughness evaluation.

Q.5.1.1.1 Upper Shelf Energy (USE)

Section IV.A.1a of Appendix G to 10 CFR Part 50 requires, in part, that the RPV beltline materials have Charpy USE in the transverse direction for base metal and along the weld for weld material of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of

Nuclear Reactor Regulation, that lower values of Charpy USE will ensure margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

By letter dated April 30, 1993, the Boiling Water Reactor Owners Group (BWROG) submitted a topical report entitled "10 CFR Part 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," to demonstrate that BWR RPVs could meet margins of safety against fracture equivalent to those required by Appendix G of the ASME Code Section XI for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the staff concluded that the topical report demonstrates that the evaluated materials have the margins of safety against fracture equivalent to Appendix G of ASME Code Section XI, in accordance with Appendix G of 10 CFR Part 50. The BWROG analysis determined that the minimum allowable Charpy USE in the transverse direction for base metal and along the weld for weld metal was 35 ft-lb.

General Electric (GE) performed an update to the USE equivalent margins analysis, which is documented in EPRI TR-113596, "BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, September 1999. The staff review and approval of EPRI TR-113596 is documented in a letter from C. I. Grimes to C. Terry dated October 18, 2001. The analysis in EPRI TR-113596 determined the reduction in the unirradiated Charpy USE resulting from neutron radiation using the methodology in RG 1.99, Revision 2. EPRI TR-113596 indicates that the percent reduction in Charpy USE for the limiting BWR/3-6 beltline plates and BWR non-Linde 80 submerged arc welds are 23.5% and 39%, respectively. For PBAPS, the predicted percent decrease of the beltline material USE values at 1/4T and 54 EFPYs was estimated using BWRVIP-74 and RG 1.99, Revision 2. The predicted percent decrease in USE for the limiting beltline plate material at the end of the license renewal period is 13% for Unit 2 and 14.5% for Unit 3; both predicted values of USE are less than the generic value of 23.5% reported in EPRI TR-113596. Similarly, the RG 1.99, Revision 2, predicted percent decrease in USE for limiting weld material (non-Linde 80 weld material at both units) at the end of license renewal period is 19.5% for both Unit 2 and Unit 3, which is less than the generic value of 39% reported in EPRI TR-113596. Therefore, the Charpy USE values at 54 EFPYs for the limiting plate and weld materials at Units 2 and 3 are greater than the minimum allowable value of 35 ft-lb, which demonstrates that the evaluated materials have the margins of safety against fracture equivalent to Appendix G of Section XI of the ASME Code, in accordance with Appendix G of 10 CFR Part 50, throughout the license renewal period.

Reference 6 contains the current fracture toughness evaluation.

Q.5.1.1.2 P-T Limit Curves

Vessel P-T limit curves for 54 EFPYs have been completed and the NRC has approved the license amendment to implement the new P-T limit curves. As part of the license amendment, the curves have been removed from the Technical Specifications and are located in Appendix D of the TRM.

The TRM contains the current P-T curves.

Q.5.1.1.3 Reactor Vessel Circumferential Weld Examination Relief (2701331-36 / T04808)

Relief has been granted from the requirements for inspection of RPV circumferential welds for the remainder of the current 40-year as well as for the 60-year period of extended operation. The relief is granted as a result of the (1) the circumferential welds satisfying the limiting conditional failure probability for circumferential welds in the BWRVIP-05 evaluation, and (2) the continued implementation of operator training and established procedures that limit the frequency of cold over-pressure events to the frequency specified in BWRVIP-05. The conditional probability of failure of the reactor pressure vessel, with no inspection of the circumferential welds, is bounded through the period of extended operation. The justification for relief is consistent with Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Guidelines.

The NRC staff used the mean RT_{NDT} value for materials to evaluate failure probability of BWR circumferential welds at 32 and 64 EFPY in the SER on BWRVIP-05 dated July 28, 1998. For PBAPS, the 54 EFPY mean RT_{NDT} values were determined to be 4.3°F and 6.6°F for Units 2 and 3, respectively. For Unit 2, the 54 EFPY fluence is 1.62E18 n/cm^2, and Cu and Ni contents are 0.056 and 0.96 wt%, respectively. For Unit 3, the 54 EFPYs fluence is 1.54E18 n/cm^2, and Cu and Ni contents are 0.102 and 0.942 wt%. These 54 EFPY values confirm that RT_{NDT} values for Units 2 and 3 are bounded by the 64 EFPYs mean RT_{NDT} value of 70.6°F used by NRC for determining the conditional failure probability of a circumferential girth weld.

The procedures and training that will be used to limit the frequency of cold over-pressure events to the number specified in the BWRVIP-05 SER for the RPV circumferential weld relief request extension, during the license renewal term, are the same as those approved for use in the initial 40-year period which used the BWRVIP-05 technical alternative.

The analysis associated with Reactor Vessel circumferential weld examination relief was projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

Reference 6 contains the current fluence and fracture toughness evaluation.

Q.5.1.1.4 Reactor Vessel Axial Weld Failure Probability

BWRVIP-05 estimated the 40-year end-of-life failure probability of a limiting Reactor Vessel axial weld, showed that it was orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds, as noted above.

The re-evaluation of the axial weld failure probability for 60 years depends on vessel ΔRT_{NDT} calculations. The NRC used Mean RT_{NDT} for the comparison. The mean RT_{NDT} values used by the NRC were determined using the neutron fluence at the clad/weld (inner) interface, and did not include a margin term. A comparison of the Mean RT_{NDT} values from the NRC report with PBAPS data shows that the NRC analysis bounds the PBAPS welds. The mean RT_{NDT} for PBAPS Units 2 and 3 is 11^0F compared to the bounding plant mean RT_{NDT} value of 91^0F. Although a conditional failure probability was not calculated, the fact that the PBAPS 54 EFPY value is less than the value the staff used leads to the conclusion that PBAPS is bounded by the NRC analysis.

The TRM contains the current P-T curves and Reference 6 contains the current fracture toughness evaluation.

Q.5.2 Metal Fatigue

The thermal and mechanical fatigue analyses of mechanical components have been identified as TLAAs for PBAPS. Specific components have been designed considering transient cycle assumptions, as listed in vendor specifications and the PBAPS UFSAR.

Q.5.2.1 Reactor Vessel Fatigue

Unit 2 and Unit 3 Reactor Vessel fatigue analyses performed for EPU assume a 60-year operating period. The effects of fatigue in the Reactor Vessel will be managed for the period of extended operation by the fatigue management program for cycle counting and fatigue usage factor tracking as described in Section Q.4.2.

This aging management program ensures that fatigue effects in Reactor Vessel pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

The fatigue evaluation of RPV closure studs is based on very conservative analysis techniques, that, in turn leads to a CUF that could exceed 1.0 during the period of extended operation.

The closure studs will be monitored by the fatigue management cycle counting and fatigue usage factor tracking program described in Section Q.4.2. As soon as the CUF value approaches 1.0, the following corrective actions will be triggered:

- Refinement of the fatigue analysis to lower the CUF to below 1.0 or
- Repair/replacement of the studs

Q.5.2.2 Reactor Vessel Internals Fatigue and Embrittlement

Q.5.2.2.1 Reactor Vessel Internals Fatigue Analyses

Original Core Shroud, Shroud Support and Jet Pump Assembly Analysis: Fatigue in these components is from both system cycles and vibrations. Vibration effects in the Core Shroud, Shroud supports, and Jet Pump assemblies were evaluated in a standard-plant analysis applicable to PBAPS. The design analysis was based on the cyclic stress criteria of ASME Section III.

Core Shroud and Jet Pump assemblies: The evaluations were extended to 60 years with a usage factor less than the allowable of 1.0.

The fatigue analyses of the Core Shroud and Jet Pump assemblies have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

Revised Core Shroud supports analysis: Fatigue analysis of the Shroud support was re-evaluated for effects of extended power uprate and 60-year plant life. The evaluation confirmed that the cumulative usage factor for the 60-year life (54 EFPY) is less than the code limit.

The usage factor at the critical Shroud support locations will be managed by the cycle counting and fatigue usage factor tracking used by the fatigue management program described in Section Q.4.2, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii). The Fatigue Management program will ensure that the fatigue effects in the shroud supports will be adequately managed for the period of extended operation.

Indications of inter-granular stress corrosion cracking (IGSCC) in the weld heat affected zones of the Unit 3 core spray piping internal to the RPV have been discovered during routine In-Vessel Visual Inspections. Repairs were installed on the piping system to mitigate some of the indications, which consisted of welded plates that span the upper T-boxes to secure the branch pipes in 1985 and U-bolt clamp hardware at the lower elbows to secure some or all of the slip joint and lower elbow welds in 1995.

During refueling outage P3R19 (2013), the entire core spray piping internal to the RPV, including piping support brackets and other components, was replaced from the RPV nozzles (N5A, N5B) to the shroud connections. All piping and welds were removed and replaced with IGSCC resistant materials (Ref. 4).

Clamps installed on the replacement core spray sparger pipes structurally replace one S1 weld and two S2 welds at each of the four sparger T-box locations, shown in Figure Q.5.1.

The RPV internal core spray system is part of the Emergency Core Cooling System (ECCS). The ECCS is a safety-related system according to the definition of Criteria I of the NRC General Design Criteria, 10 CFR 50, Appendix A.

In accordance with reference 3, the service life of the RPV internal core spray piping and hardware is 40 effective full power years. Therefore, these components remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

Q.5.2.2.2 Reactor Vessel Internals Embrittlement Analyses

The Reactor Vessel Internals embrittlement analyses affect the core shroud and top guides.

Shroud: An evaluation was performed for EPU and found that the expected 60-year fluence on the shroud exceeds threshold value for irradiation assisted stress corrosion cracking (IASCC). PBAPS follows BWRVIP guidelines that incorporate consideration of the fluence exceeding threshold value. The recommended inspections are based on component configuration and field experience. Plant-specific fluence values are used to evaluate the detected flaw evaluations and determine the inspection interval in accordance with BWRVIP-76.

Top Guide: An evaluation was performed for EPU, and although the expected 60-year fluence on the top guide exceeds threshold value for irradiation assisted stress corrosion cracking (IASCC), critical locations in the top guide at PBAPS have low tensile stresses and thus are exempt from inspection under the BWRVIP-26 guidelines. That is, at these low stresses, a fracture is not a concern, and embrittlement is therefore not a threat to the intended function.

The existing analyses of the effects of embrittlement in core shrouds and top guides have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

Q.5.2.2.3 Effect of Fatigue and Embrittlement on End-of-Life Reflood Thermal Shock Analysis

Radiation embrittlement and fatigue usage may affect the ability of certain internals, particularly the core shroud support plate, to withstand an end-of-life reflood thermal shock following a recirculation line break.

The existing analyses of the effects of fatigue and embrittlement on end-of-life reflood thermal shock analyses of reactor internals have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

Q.5.2.3 Piping and Component Fatigue and Thermal Cycles

Q.5.2.3.1 Fatigue Analyses of Group I Primary System Piping

All Group I piping was originally designed in accordance with USAS B31.1, 1967 Edition. The Recirculation and RHR piping in Group 1 was replaced under the GL 88-01 IGSCC Correction Program. The replacement piping was analyzed to ASME Section III Class 1 rules. ASME Section III requires an explicit fatigue analysis for Class 1 components. USAS B.31.1 does not require an explicit fatigue analysis. However, a simplified fatigue analysis was developed to estimate the current usage factors from operating data for the fatigue management cycle counting and fatigue usage factor tracking program.

The effects of fatigue in Group I primary system piping will be managed for the period of extended operation by the fatigue management cycle counting and fatigue usage factor tracking program, as part of the Fatigue Management Activities, discussed in Section Q.4.2. This aging management activity will ensure that fatigue effects in pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

Q.5.2.3.2 Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction in Group II and III Piping and Components

Group II and III Piping: Thermal cycle count is a consideration in all the codes associated with the design of Group II and III piping (e.g., USAS or ANSI B31.1). The applicable piping codes require the use of a stress range reduction factor in the evaluation of calculated stresses due to thermal expansion. The reduction factor is based on the anticipated number of equivalent full temperature cycles over the total number of years the plant is expected to be in operation.

The number of thermal cycles assumed for design of Group II and III piping has been evaluated and the existing stress range reduction factor remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Q.5.2.3.3 Design of the RHR System for a Finite Number of Cycles

Group I RHR piping inside the Drywell that was analyzed to ASME III Class 1 rules is discussed in Section Q.5.2.3.1.

Group II RHR piping and some Group I RHR piping is designed to a USAS or ANSI B31.1 allowable secondary stress range determined by a finite number of equivalent full-temperature thermal cycles, without a detailed fatigue analysis.

The ANSI B31.1 threshold number for applying a reduction factor for RHR valves is 7,000 equivalent full-temperature cycles. The total number of cycles assumed for the original 40-year plant life is, conservatively, less than 1,000. For the period of extended operation, the number of thermal cycles for piping analyses would be proportionately increased to 1,500, which is still significantly less than the 7,000-cycle threshold. The code stress range reduction factor therefore remains at 1.0 and is not affected by extending the operating period to 60 years

The RHR valve and piping thermal cycle analyses have been evaluated and remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Q.5.2.4 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

Generic Safety Issue (GSI) 190 was identified by the NRC because of concerns about potential effects of reactor water environments on component fatigue life during the period of extended operation. The GSI was closed in December 1999 because the NRC concluded that environmental effects have a negligible effect on core damage frequency; however, license renewal applicants need to address the effects of the coolant environment on component fatigue life.

To evaluate the impact of reactor water environment on the fatigue life of components, plant-specific calculations were performed for the locations identified in NUREG/CR-6260 for the older vintage BWR plant. For each of these locations, detailed environmental fatigue calculations were performed using the appropriate F_{en} relationships from NUREG/CR-0609, Revision 0. These calculations were performed prior to entry into the period of extended operation, and appropriate corrective action was not required since the resulting CUF values did not exceed 1.0.

Exelon reserves the right to modify this position in the future based on the results of industry activities currently underway, as well as based on the results of any other methodology improvements that may be made associated with environmental fatigue. It is understood that any such modifications will be subject to NRC approval prior to implementation at PBAPS.

Q.5.3 Environmental Qualification Of Electrical Equipment

Electrical equipment included in the PBAPS Environmental Qualification Program which has a specified qualified life of at least 40 years involves time-limited aging analyses for license renewal. The aging effects of this equipment will be managed in the Environmental Qualification Program discussed in Q.4.1, Environmental Qualification Activities, in accordance with the requirements of 10CFR54.21(c)(1)(iii).

Q.5.4 Containment Fatigue

Subsequent to the original design, elements of the PBAPS containment were re-analyzed in response to discoveries of unevaluated loads due to design basis events and Safety Relief Valve (SRV) discharge. This re-evaluation was in two parts: Generic analyses applicable to each of the several classes of BWR containments, and plant-unique analyses (PUA). The scope of the analyses included the Tori, the Drywell-to-Torus vents (Torus vents), SRV discharge piping, other torus-attached piping and its penetrations, and the Torus vent bellows.

Q.5.4.1 Fatigue Analyses of Containment Pressure Boundaries: Analysis of Tori, Torus Vents, and Torus Penetrations

For low usage factor locations (40-year CUF < 0.4) the PBAPS new loads analyses of Tori, Torus vents, and Torus penetrations have been evaluated and determined to remain valid for the extended period of operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

For higher usage factor locations in the analyses of Tori, Torus vents, and Torus penetrations (40-year CUF ≥ 0.4) the effects of fatigue will be managed for the period of extended operation by the fatigue management cycle counting and fatigue usage factor tracking program described in Section Q.4.2.

The Fatigue Management activities will ensure that fatigue effects in Reactor Vessel pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

Q.5.4.2 Fatigue Analysis of SRV Discharge Lines and External Torus-Attached Piping

SRV discharge lines and external Torus-attached piping were analyzed separately from the Tori and Torus vents. The PBAPS analysis included the SRV lines, all piping and branch lines attached to the Tori, pipe supports, valves, flanges, equipment nozzles, and equipment anchors.

The fatigue analyses of SRV discharge lines and external Torus-attached piping have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Q.5.4.3 Expansion Joint and Bellows Fatigue Analyses - Drywell to Torus Vent Bellows

The predicted fatigue usage factors for the Drywell to Torus vent bellows for the period of extended operation are negligible.

The PBAPS new loads fatigue analyses of the Drywell-to-Torus vent bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Q.5.4.4 Expansion Joint and Bellows Fatigue Analyses - Containment Process Penetration Bellows

The only Containment process piping expansion joints subject to significant thermal expansion and contraction are those between the Drywell shell penetrations and process piping. These are designed for a stated number of operating and thermal cycles.

The thermal cycle designs of PBAPS Containment process penetration bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Q.5.5 Metal Corrosion Allowances

Q.5.5.1 Reactor Vessel Corrosion Main Steam Nozzle Cladding Removal Allowance

The original Reactor Vessel corrosion allowances were conservative values intended to encompass 40 years of operation but without reliance on a particular corrosion rate. The original allowances: therefore, do not depend on the 40-year design life. However, a subsequent calculation to justify removal of the Main Steam nozzle cladding used a time-dependent corrosion rate and is thereby a TLAA.

The Reactor Vessel corrosion allowances have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

Q.5.6 Inservice Flaw Growth Analyses That Demonstrate Structural Integrity For 40 Years

Two flaw dispositions were identified that include TLAAs.

Q.5.6.1 Generic Letter 81-11 Crack Growth Analysis to Demonstrate Conformance to the Intent of NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking”

The PBAPS Control Rod Drive Hydraulic System return nozzles were capped. Therefore, cracking of the Feedwater nozzles is the only aging effect that applies to PBAPS. Cracking of the Feedwater nozzles was addressed by installing triple thermal sleeves with double-piston-ring seals, by removing the vessel clad from the nozzles, by installing improved low-flow Feedwater controllers, and by adopting and maintaining an augmented inspection program to detect incipient problems currently using the NRC-approved BWR Owner’s Group (BWROG) inspection and management methods.

The fracture mechanics evaluations, which support the validity of the current examination methods, are not TLAAs. However, the effects of the cracking phenomena must be managed to ensure the continued validity of the assumptions of fatigue analyses for the reactor vessel, which are TLAAs.

The aging effect is adequately addressed by the modifications already installed and by the inspection program already in place. No enhancements are required.

Any remaining or recurring effects of rapid-thermal-cycle damage at Feedwater nozzle inner blend radii will be managed for the period of extended operation by the Reactor Vessel and Internals Inspection Program described in Section Q.2.7. This aging management activity includes specific requirements for these nozzles and for this issue.

This program will ensure that any effects will be adequately detected, managed, and controlled, within the limits of the supporting fracture mechanics analyses, for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

Q.5.6.2 Fracture Mechanics of As-forged Laminar Tear in a Unit 3 Main Steam Elbow

Preservice inspection discovered an as-forged laminar tear in a Unit 3 Main Steam elbow near weld 1-B-3BC-LDO. The original disposition included a fatigue analysis.

The fatigue analysis for the as-forged laminar tear has been evaluated and remains valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

Q.5.7 Crane Load Cycle Limit

The following cranes have load cycle assumptions that result in the fatigue analyses, considered by NRC staff a TLAA:

- Reactor Building overhead bridge cranes
- Turbine Hall cranes
- Emergency Diesel Generator bridge cranes
- Circulating Water Pump Structure gantry crane

The load cycles for these cranes were evaluated for the period of extended operation. For each crane, the actual usage over its projected life through the period of extended operation will be less than the analyzed number of cycles. The cranes will continue to perform their intended function throughout the period of extended operation.

Therefore, the analysis associated with load cycle limit for the cranes, remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Q.6 References

(1) BWRVIP-05, EPRI Report TR-105697, "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)." For the Boiling Water Reactor Owners Group (Proprietary), September 28, 1995, with supplementing letters of June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998.

(2) BWRVIP-26, EPRI Report TR-107285, "BWR Vessel and Internals Project: BWR Top Guide Inspection and Flaw Evaluation Guidelines", December 1996.

(3) GEH Core Spray Line Replacement Design Specification, 26A8106, Rev. 3 (M-1-B-464).

(4) ECR 10-00279 Rev. 1, "U3 Core Spray In-Vessel Piping Replacement."

(5) NEDC-33566P, Safety Analysis Report for Exelon Peach Bottom Atomic Power Station, Units 2 and 3, Constant Pressure Power Uprate, Revision 0, September 2012.

(6) GE Hitachi Nuclear Energy, "PBAPS Units 2 & 3 TPO Task T0301 – RPV Fracture Toughness Evaluation," GEH, Wilmington, NC, 00N5577, October 2016 (PEAM-MUR-0301).

Q.7 Newly Identified Items (10 CFR 54.37(b))

After the renewed license is issued, the UFSAR update required by 10 CFR 50.71(e) must include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with § 54.21. This UFSAR update must describe how the effects of aging will be managed such that the intended function(s) in § 54.4(b) will be effectively maintained during the period of extended operation.

Q.7.1 Newly Identified Structures, Systems and Components (SSC)

No.	UFSAR Revision Date	SSC Description	Aging Management Review (AMR) Conclusion	Aging Management Program(s)
1.	04/06/17	The Refueling Water Storage Tank (RWST) was not in scope for the first license renewal. The Extended Power Uprate (EPU) project eliminated reliance on Containment Accident Pressure (CAP) in Emergency Core Cooling System pump net positive suction head evaluations. In order to support elimination of reliance on CAP credit, the Condensate Storage Tanks (CSTs) are utilized as the sole suction source for High Pressure Coolant Injection/Reactor Core Isolation Cooling systems during Station Blackout (SBO), Anticipated Transient Without Scram (ATWS), and Appendix R events. The volume of the RWST is required to supplement the CST for the Appendix	Components added to scope as a result of this change: • Heating Coil • Pipe • Pump Casings • Refueling Water Storage Tank • Refueling Water Storage Tank Nozzles • Restricting Orifice • Tubing • Valve Bodies Materials of construction: • Carbon Steel • Cast Iron • Stainless Steel Environments: • Auxiliary Steam (Internal) • Buried (External)	The Demineralized Water and Condensate/Refueling Water Storage Tank Chemistry Activities (Q.1.4) AMP is revised to periodically monitor RWST water to assure that purity is maintained within pre-established limits. If parameter limits are exceeded, corrective actions are taken to restore parameters within acceptable limits.

No.	UFSAR Revision Date	SSC Description	Aging Management Review (AMR) Conclusion	Aging Management Program(s)
		R and ATWS events. The RWST and SSCs necessary to establish a flowpath to transfer the supplemental volume to the CST are credited for Appendix R and ATWS events and are added to scope for license renewal with an intended function of Pressure Boundary.	• Condensate Storage Water (Internal/External) • Outdoor (External) • Sheltered (External) Aging Effects for the material/environment combinations: • Cracking of stainless steel in Auxiliary Steam, Condensate Storage Water, Outdoor, and Buried environments • Loss of material in stainless steel in Auxiliary Steam, Condensate Storage Water, Outdoor, and Buried environments • Loss of material in carbon steel/cast iron in Condensate Storage Water, Outdoor, and Buried environments	The Outdoor, Buried and Submerged Component Inspection Activities (Q.2.5) AMP is revised to include periodic visual inspection of the outdoor Refueling Water Storage and Transfer System piping and valves, the external surfaces of the RWST, and the piping insulation jacketing at the RWST. Revised program activities also include visual inspection of buried Refueling Water Storage and Transfer System piping whenever it is uncovered during excavation.
2.	04/06/17	The torus dewatering tank dike was not in scope for the first license renewal. As a result of EPU, the torus dewatering tank dike must provide structural support for piping credited with establishing a flowpath from the RWST to the Unit 3 CST. Since the torus dewatering tank dike is now credited for	Components added to scope as a result of this change: • Reinforce Concrete: Above-grade exterior (accessible areas) • Reinforced Concrete: Above-grade exterior (inaccessible areas) • Reinforced Concrete: Below-	The torus dewatering tank dike is added to the scope of the Maintenance Rule Structural Monitoring Program (Q.1.16).

No.	UFSAR Revision Date	SSC Description	Aging Management Review (AMR) Conclusion	Aging Management Program(s)
		Appendix R and ATWS events, it is added to scope for license renewal with an intended function of Structural Support to nonsafety-related components.	grade exterior (inaccessible areas) Materials of construction: • Concrete Environments: • Buried (External) • Outdoor (External) Aging Effects for the material/environment combinations: • Cracking, loss of material, change in material properties of concrete in an Outdoor environment	
3.	04/06/17	The following hoists were not in scope for the first license renewal. As a result of review of the current revision of engineering analysis, M-25-102, Overhead Handling Systems Review Final Report, it was identified that the following nonsafety-related hoists are above or nearby to safety-related SSCs, and therefore should have been identified as in scope for license renewal with a 10 CFR54.4(a)(2) structural intended function. - Radwaste Building Demin Hoist - Dewatering Building Coffing Hoist - U3 1 Ton Crane Over Repair Area	There are no changes to Aging Management Review for these hoists as shown in LRA Table 3.3.18. Components added to scope as a result of this change: The three hoists being added are included under the Component Group – "Other Cranes and Hoists" with a Component Intended Function of "Structural Support to Non-S/R Components". Materials of construction: The in scope structural components associated with	The following hoists are added to the Crane Inspection Activities program (Q.1.14). - Radwaste Building Demin Hoist - Dewatering Building Coffing Hoist - U3 1 Ton Crane Over Repair Area The Crane Inspection Activities program requires periodic inspection of in scope components for

No.	UFSAR Revision Date	SSC Description	Aging Management Review (AMR) Conclusion	Aging Management Program(s)
		The list of cranes and hoists that were in scope for license renewal was provided in RAI 2.3.3.18-1. At the time of the first license renewal application, these three hoists were installed in the plant, however M-25-102 did not correctly identify that they were above or nearby safety-related SSCs.	these hoists are carbon steel, consistent with the materials for the Other Cranes and Hoists. Environment: All three hoists are in a "Sheltered" environment, consistent with the environment for the Other Cranes and Hoists. Aging Effects for the material/environment combinations: Loss of Material, consistent with the aging effects required to be managed for the Other Cranes and Hoists.	loss of material. Review of inspection activities for these hoists indicates that periodic inspections meeting the Crane Inspection Activities program requirements have been, and are continuing to be performed, in accordance with program procedures.

Q.7.2 Newly Identified Time Limited Aging Analyses (TLAA)

No.	UFSAR Revision Date	TLAA Description	TLAA Evaluation	TLAA Disposition
1.	04/06/17	Unit 3 Jet Pump Riser Repair Clamps: During the Fall 1997 refueling outage for Unit 3, crack indications were detected in the heat-affected zones of the jet pump riser elbow-to-thermal sleeve welds	In order to determine if this fluence assumption will remain valid through 60 years of operation, the neutron fluence was projected for 54 EFPY using the RAMA fluence methodology. The effects of the MELLLA+	The 54 EFPY fluence value at the Unit 3 jet pump riser clamp location was determined to be approximately 1.40E+17 n/cm^2,

No.	UFSAR Revision Date	TLAA Description	TLAA Evaluation	TLAA Disposition
		for Jet Pump numbers 01/02 and 13/14. A mechanical clamping system was designed to structurally replace these welds and these clamps were installed in 1998. Since these clamshell-style clamps use bolts to maintain the proper clamping force, preload loss due to irradiation induced stress relaxation was a design consideration. The design specification provided a conservative estimate of the neutron fluence at the riser pipe clamp location to be 2.5E+19 n/cm^2 during the 17 year planned service life of the clamp. Since the neutron fluence value was estimated through the end of the initial 40 years of operation, the design analysis has been identified as a TLAA that requires evaluation for the period of extended operation.	operation and the projected 1.7 percent MUR power uprate were considered in this fluence projection. The 54 EFPY fluence value at the Unit 3 jet pump riser clamp location was determined to be approximately 1.40E+17 n/cm^2, which is less than the 2.5E+19 n/cm^2 fluence value assumed in the design specification for the clamp.	which is less than the 2.5E+19 n/cm^2 fluence value assumed in the design specification for the clamp. Therefore, the riser clamp design analysis based upon the design specification remains valid through the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).