

APPENDIX R SECOND LICENSE RENEWAL UFSAR SUPPLEMENT

R.1.0 INTRODUCTION

This appendix contains the UFSAR Supplement as required by 10 CFR 54.21(d) for the Peach Bottom Second License Renewal Application (SLRA). The NRC issued the Final SER (dated February 2020) that provides their safety evaluation of the Peach Bottom SLRA. This appendix, which includes the following sections (corresponding to the respective SLRA Appendix A sections), comprises the FSAR supplement:

- Section R.1.1 contains a listing of the aging management programs that correspond to NUREG-2191 Chapter XI programs, including the status of the programs at the time the Second License Renewal Application was submitted.
- Section R.1.2 contains a listing of the plant-specific aging management programs, including the status of the programs at the time the Second License Renewal Application was submitted.
- Section R.1.3 contains a listing of aging management programs that correspond to NUREG-2191 Chapter X programs associated with Time-Limited Aging Analyses, including the status of the programs at the time the Second License Renewal Application was submitted.
- Section R.1.4 contains a listing of the Time-Limited Aging Analyses summaries (TLAAs).
- Section R.1.5 contains a discussion of the Quality Assurance Program and Administrative Controls.
- Section R.2 contains a summarized description of the aging management programs.
- Section R.2.1 contains a summarized description of the NUREG-2191 Chapter XI programs for managing the effects of aging.
- Section R.2.2 contains a summarized description of the plant-specific programs for managing the effects of aging.
- Section R.3 contains a summarized description of the NUREG-2191 Chapter X programs that support the TLAAs.
- Section R.4 contains a summarized description of the TLAAs applicable to the second period of extended operation.
- Section R.5 contains the Second License Renewal Commitment List.

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

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The integrated plant assessment for license renewal identified new and existing aging management programs necessary to provide reasonable assurance that systems, structures, and components within the scope of license renewal will continue to perform their intended functions consistent with the Current Licensing Basis (CLB) for the period of extended operation. The second period of extended operation is defined as 20 years from the unit's current operating license expiration date.

Activities Credited for Managing Aging in the Second Renewal Term

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 for the second period of extended operation.

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

- Existing Activity: An activity in existence prior to second license renewal approval that will continue to be implemented during the second period of extended operation. (Although some activities were existing at the time of SLRA issue, these had to be modified due to Requests for Additional Information (RAIs) and SLRA Supplements. To maintain the same numbering scheme, these are still considered existing activities rather than enhanced activity.)
- Enhanced Activity: An activity in existence prior to second license renewal approval that will be modified during the second period of extended operation. Enhancements were implemented as discussed in this Appendix.
- New Activity: An activity that did not exist prior to second license renewal approval, which will manage aging during the second period of extended 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.

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 special treatment programs as defined in 10 CFR 50.69, for components that have a Risk-Informed Safety Classification (RISC) RISC-3 or RISC-4. When Alternative Treatment is applied, aging effects will continue to be managed under License Renewal Aging Management Programs (AMPs).

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R.1.1 NUREG-2191 Chapter XI Aging Management Programs

The NUREG-2191 Chapter XI Aging Management Programs (AMPs) are described in the following sections. The AMPs are either consistent with generally accepted industry methods as discussed in NUREG-2191 or require enhancements.

The following list reflects the status of these programs at the time of the Second License Renewal Application (SLRA) submittal. Commitments for program additions and enhancements are identified in the Appendix R.5 Second License Renewal Commitment List.

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (Section R.2.1.1) [Existing]
2. Water Chemistry (Section R.2.1.2) [Existing]
3. Reactor Head Closure Stud Bolting (Section R.2.1.3) [Existing]
4. BWR Vessel ID Attachment Welds (Section R.2.1.4) [Existing]
5. BWR Stress Corrosion Cracking (Section R.2.1.5) [Existing]
6. BWR Penetrations (Section R.2.1.6) [Existing]
7. BWR Vessel Internals (Section R.2.1.7) [Existing - Requires Enhancement]
8. Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section R.2.1.8) [New]
9. Flow-Accelerated Corrosion (Section R.2.1.9) [Existing - Requires Enhancement]
10. Bolting Integrity (Section R.2.1.10) [Existing - Requires Enhancement]
11. Open-Cycle Cooling Water System (Section R.2.1.11) [Existing - Requires Enhancement]
12. Closed Treated Water Systems (Section R.2.1.12) [Existing - Requires Enhancement]
13. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section R.2.1.13) [Existing - Requires Enhancement]
14. Compressed Air Monitoring (Section R.2.1.14) [Existing - Requires Enhancement]
15. BWR Reactor Water Cleanup System (Section R.2.1.15) [Existing]

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16. Fire Protection (Section R.2.1.16) [Existing - Requires Enhancement]
17. Fire Water System (Section R.2.1.17) [Existing - Requires Enhancement]
18. Outdoor and Large Atmospheric Metallic Storage Tanks (Section R.2.1.18) [Existing - Requires Enhancement]
19. Fuel Oil Chemistry (Section R.2.1.19) [Existing - Requires Enhancement]
20. Reactor Vessel Material Surveillance (Section R.2.1.20) [Existing - Requires Enhancement]
21. One-Time Inspection (Section R.2.1.21) [New]
22. Selective Leaching (Section R.2.1.22) [New]
23. ASME Code Class 1 Small-Bore Piping (Section R.2.1.23) [New]
24. External Surfaces Monitoring of Mechanical Components (Section R.2.1.24) [New]
25. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (Section R.2.1.25) [New]
26. Lubricating Oil Analysis (Section R.2.1.26) [Existing]
27. Monitoring of Neutron-Absorbing Materials Other Than Boraflex (Section R.2.1.27) [Existing]
28. Buried and Underground Piping and Tanks (Section R.2.1.28) [Existing - Requires Enhancement]
29. Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (Section R.2.1.29) [New]
30. ASME Section XI, Subsection IWE (Section R.2.1.30) [Existing - Requires Enhancement]
31. ASME Section XI, Subsection IWF (Section R.2.1.31) [Existing - Requires Enhancement]
32. 10 CFR Part 50, Appendix J (Section R.2.1.32) [Existing]
33. Masonry Walls (Section R.2.1.33) [Existing - Requires Enhancement]
34. Structures Monitoring (Section R.2.1.34) [Existing - Requires Enhancement]

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35. Inspection of Water-Control Structures Associated with Nuclear Power Plants (Section R.2.1.35) [Existing - Requires Enhancement]
36. Protective Coating Monitoring and Maintenance (Section R.2.1.36) [Existing - Requires Enhancement]
37. Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section R.2.1.37) [Existing - Requires Enhancement]
38. Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section R.2.1.38) [Existing - Requires Enhancement]
39. Electrical Insulation for Inaccessible Medium-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section R.2.1.39) [Existing - Requires Enhancement]
40. Electrical Insulation for Inaccessible Instrument and Control Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section R.2.1.40) [New]
41. Electrical Insulation for Inaccessible Low-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section R.2.1.41) [New]
42. Metal Enclosed Bus (Section R.2.1.42) [New]
43. Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section R.2.1.43) [New]

R.1.2 Plant-Specific Aging Management Programs

The plant-specific aging management programs are described in the following sections. The following list reflects the status of these programs at the time of the Second License Renewal Application (SLRA) submittal. Commitments for program additions and enhancements are identified in Appendix R.5 Second License Renewal Commitment List.

1. Wooden Pole (Section R.2.2.1) [Existing - Requires Enhancement]

R.1.3 NUREG-2191 Chapter X Aging Management Programs

The NUREG-2191 Chapter X Aging Management Programs (AMP) associated with Time-Limited Aging Analyses are described in the following sections. The AMPs are either consistent with generally accepted industry methods as discussed in NUREG-2191 Chapter X or require enhancements. The following list reflects the status of these programs at the time of the Second License Renewal Application (SLRA) submittal. Commitments for program additions and enhancements are identified in Appendix R.5 Second License Renewal Commitment List.

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1. Fatigue Monitoring (Section R.3.1.1) [Existing - Requires Enhancement]
2. Neutron Fluence Monitoring (Section R.3.1.2) [Existing - Requires Enhancement]
3. Environmental Qualification of Electric Equipment (Section R.3.1.3) [Existing - Requires Enhancement]

R.1.4 Time-Limited Aging Analyses

Summaries of the Time-Limited Aging Analyses applicable to the second period of extended operation are included in the following sections:

1. Reactor Vessel and Internals Neutron Embrittlement Analyses (Section R.4.2.1)
2. Reactor Vessel Neutron Fluence Analyses (Section R.4.2.1.1)
3. Reactor Vessel Internals Neutron Fluence Analyses (Section R.4.2.1.2)
4. Reactor Vessel Upper-Shelf Energy (USE) Analyses (Section R.4.2.2)
5. Reactor Vessel Adjusted Reference Temperature (ART) Analyses (Section R.4.2.3)
6. Reactor Vessel Pressure – Temperature Limits (Section R.4.2.4)
7. Reactor Vessel Circumferential Weld Failure Probability Analyses (Section R.4.2.5)
8. Reactor Vessel Axial Weld Failure Probability Analyses (Section R.4.2.6)
9. Reactor Vessel Reflood Thermal Shock Analysis (Section R.4.2.7)
10. Core Shroud Reflood Thermal Shock Analysis (Section R.4.2.8)
11. Core Plate Rim Hold-Down Bolt Loss of Preload Analysis (Section R.4.2.9)
12. Jet Pump Slip Joint Repair Clamp Loss of Preload Analysis (Section R.4.2.10)
13. Jet Pump Auxiliary Spring Wedge Assembly Loss of Preload Analysis (Section R.4.2.11)
14. Jet Pump Riser Repair Clamp Loss of Preload Analysis (Section R.4.2.12)
15. Replacement Core Plate Extended Life Plug Irradiation-Enhanced Stress Relaxation Analysis (Section R.4.2.13)
16. First License Renewal Application Core Shroud IASCC and Embrittlement Analysis (Section R.4.2.14)

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17. Unit 3 Core Spray Replacement Piping Bolting Loss of Preload Evaluation (Section R.4.2.15)
18. Transient Cycle and Cumulative Fatigue Usage Projections for 80 Years (Section R.4.3.1)
19. ASME Section III, Class 1 Fatigue Analyses (Section R.4.3.2)
20. ASME Section III, Class 1 Fatigue Waivers (Section R.4.3.3)
21. ASME Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Analyses (Section R.4.3.4)
22. Environmental Fatigue Analyses for RPV and Class 1 Piping (Section R.4.3.5)
23. Reactor Vessel Internals Fatigue Analyses (Section R.4.3.6)
24. Generic BWR Fatigue Analyses for Various Reactor Vessel Internal Components (Section R.4.3.6.1)
25. Generic BWR Fatigue Analyses for the Core Shroud (Section R.4.3.6.2)
26. Core Shroud Support Fatigue Analyses Reevaluation (Section R.4.3.6.3)
27. Jet Pump Diffuser/Core Shroud Support Plate Fatigue Analysis (Section R.4.3.6.4)
28. Replacement Steam Dryer Stress Report and Fatigue Evaluation (Section R.4.3.6.5)
29. High-Energy Line Break (HELB) Analyses Based Upon Cumulative Fatigue Usage (Section R.4.3.7)
30. Inservice 60-Year RPV Closure Head Weld Flaw Analyses (R.4.3.8)
31. Environmental Qualification of Electric Equipment (Section R.4.4.1)
32. Concrete Containment Tendon Prestress Analyses (Section R.4.5.1)
33. Primary Containment Structures, Penetrations, and Associated Components with Fatigue Analyses (Section R.4.6.1)
35. Containment Process Line Penetration Bellows (Section R.4.6.2)
36. Cranes Cyclic Loading Analyses (Section R.4.7.1)
37. Reactor Vessel Main Steam Nozzle Clad Removal Corrosion Allowance (Section R.4.7.2)
38. 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" (Section R.4.7.3)

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39. Fracture Analysis of ISI-Reportable Indications for Group I Piping: As-forged Laminar Tear in a Unit 3 Main Steam Elbow Near Weld 1-B-3BC-LDO Discovered During Preservice UT (Section R.4.7.4)
40. Unit 3 Core Spray Replacement Piping Fatigue and Leakage Assessment (Section R.4.7.5)

R.1.5 Quality Assurance Program and Administrative Controls

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2, "Quality Assurance For Aging Management Programs (Branch Technical Position IQMB-1)" of NUREG-2192. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and nonsafety-related systems, structures, and components (SSCs) that are subject to Aging Management Review (AMR). In many cases, existing activities were found adequate for managing aging effects during the second period of extended operation.

R.1.6 Operating Experience

Operating experience from plant-specific and industry sources is captured and systematically reviewed on an ongoing basis in accordance with the quality assurance program, which meets the requirements of 10 CFR Part 50, Appendix B, and the operating experience program, which meets the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff." The operating experience program interfaces with and relies on active participation in the INPO operating experience program, as endorsed by the NRC. In accordance with these programs, all incoming operating experience items are screened to determine whether they may involve age-related degradation or aging management impacts. Research and development is also reviewed. Items so identified are further evaluated and the AMPs are either enhanced or new AMPs are developed, as appropriate, when it is determined through these evaluations that the effects of aging may not be adequately managed. Training on age-related degradation and aging management is provided to those personnel responsible for implementing the AMPs and to those who may submit, screen, assign, evaluate, or otherwise process plant-specific and industry operating experience. Plant-specific operating experience associated with aging management and age-related degradation is reported to the industry in accordance with guidelines established in the operating experience program.

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R.2.0 AGING MANAGEMENT PROGRAMS

R.2.1 NUREG-2191 Chapter XI Aging Management Programs

This section provides summaries of the NUREG-2191 programs credited for managing the effects of aging.

R.2.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (3953023-01)

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is an existing condition monitoring program that consists of periodic volumetric, surface, and visual examinations of ASME Class 1, 2, and 3 components including welds, pump casings, valve bodies, integral attachments, and pressure-retaining bolting for assessment, identification of signs of degradation, and establishment of corrective actions. The examinations are implemented in accordance with 10 CFR 50.55a and ASME Code, Section XI Subsections IWB, IWC, and IWD. The program includes augmented inservice inspection requirements for periodic examination of the reactor vessel feedwater nozzles in accordance with the staff-approved recommendations provided within BWR Owners Group (BWROG) Licensing Topical Report, "Alternate BWR Feedwater Nozzle Inspection Requirements," GE-NE-523-A71-0594-A, Revision 1, May 2000. These activities include examinations, evaluations, monitoring, and trending of results to confirm that effects of cracking, loss of material, loss of fracture toughness, and loss of preload for pressure-retaining bolting are managed effectively during the second period of extended operation.

R.2.1.2 Water Chemistry (3953023-02)

The Water Chemistry program is an existing program that mitigates aging effects of loss of material due to corrosion, cracking due to SCC, and related mechanisms, and reduction of heat transfer due to fouling in components exposed to a reactor coolant, steam, or treated water environment. Chemistry programs are used to control water chemistry for parameters such as conductivity, chloride, and sulfate that accelerate corrosion. The Water Chemistry program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits, based on the industry recognized guidelines of the Boiling Water Reactor Vessel and Internals Project (BWRVIP-190, Revision 1, Electric Power Research Institute - 3002002623). In addition, the Water Chemistry program is credited for mitigating loss of material and cracking for components exposed to sodium pentaborate and auxiliary steam environments.

R.2.1.3 Reactor Head Closure Stud Bolting (3953023-03)

The Reactor Head Closure Stud Bolting aging management program is an existing condition monitoring and preventive program that manages reactor head closure studs, flange threads, and associated nuts, washers, and bushings, for cracking and loss of material. The program is implemented

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through station procedures based on the examination requirements specified in ASME Code, Section XI, Subsection IWB, Table IWB-2500-1 and preventive measures to mitigate cracking as delineated in NRC Regulatory Guide 1.65, Revision 1, with the exception that existing stud bolting components have a measured yield strength greater than or equal to 150 ksi and an ultimate tensile stress greater than or equal to 170 ksi. Also, potential replacement stud bolting components in the warehouse have measured yield strength greater than or equal to 150 ksi.

R.2.1.4 BWR Vessel ID Attachment Welds (3953023-04)

The BWR Vessel ID Attachment Welds aging management program is an existing condition monitoring program that manages cracking of the reactor vessel interior attachment welds. This program relies on visual examinations to detect cracking. The examination scope, frequencies, and methods are in accordance with ASME Code, Section XI, Table IWB-2500-1, Examination Category B-N-2, and BWRVIP-48-A. The scope of the examinations is expanded when flaws are detected.

Any indications are evaluated in accordance with ASME Code, Section XI, or the guidance in BWRVIP-48-A. Crack growth evaluations follow the guidance in BWRVIP-14-A, BWRVIP-59-A, or BWRVIP-60-A, as appropriate. The acceptance criteria are in BWRVIP-48-A and ASME Code, Section XI, Subarticle IWB-3520. Repair and replacement activities are conducted in accordance with BWRVIP-52-A.

R.2.1.5 BWR Stress Corrosion Cracking (3953023-05)

The BWR Stress Corrosion Cracking aging management program is an existing condition monitoring and mitigative program that manages intergranular stress corrosion cracking (IGSCC) for all BWR piping and piping welds made of austenitic stainless steel and nickel based alloy that are 4 inches or larger in diameter containing reactor coolant at a temperature above 200 degrees F during power operation, regardless of code classification, with the exception of reactor water cleanup system piping that is outboard of the second (outboard) primary containment isolation valve, that is managed by the BWR Reactor Water Cleanup System (R.2.1.15) aging management program.

The program includes periodic volumetric examinations to detect and manage IGSCC in accordance with NRC GL 88-01. The extent and schedule of inspection described in GL 88-01 are modified in accordance with the inspection guidance in staff-approved BWRVIP-75-A. Welds classified as IGSCC Category A may be inspected at a frequency in accordance with ASME Section XI, including the Code Case N-716-1 Risk Informed Inspection (RI ISI) program. The program includes the staff-approved positions delineated in NUREG-0313, Revision 2, and GL 88-01 and its Supplement 1 regarding selection of IGSCC resistant materials, solution heat treatment and stress improvement processes, water chemistry, weld overlay reinforcement, partial replacement, clamping devices, crack characterization and repair criteria, inspection methods and personnel, inspection schedules, sample expansion,

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leakage detection, and reporting requirements.

R.2.1.6 BWR Penetrations (3953023-06)

The BWR Penetrations aging management program is an existing condition monitoring program that manages the effects of cracking due to cyclic loading or stress corrosion cracking of BWR instrumentation penetrations, CRD housing and incore-monitoring housing penetrations, and the SLC/Core Plate dP nozzle exposed to reactor coolant by performing inspections and flaw evaluations. In addition to the requirements of ASME Code, Section XI, Subsection IWB, the BWR Penetrations program incorporates the inspection and flaw evaluation recommendations of BWRVIP-49-A, "Instrument Penetration Inspection and Flaw Evaluation Guidelines," BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines," BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate dP Inspection and Flaw Evaluation Guidelines," and the water chemistry recommendations described in the Water Chemistry (R.2.1.2) program. The examination categories include volumetric, surface, and visual examination methods.

R.2.1.7 BWR Vessel Internals (3953023-07)

The BWR Vessel Internals aging management program is an existing condition monitoring and mitigative program that includes inspections and flaw evaluations in conformance with the guidelines of applicable staff-approved BWRVIP documents, and provides reasonable assurance of the long-term integrity and safe operation of BWR vessel internal components that are fabricated of nickel alloy and stainless steel (including martensitic stainless steel (not installed in PBAPS reactor vessel internals), cast stainless steel and associated welds).

The program manages the effects of cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), or irradiation assisted stress corrosion cracking (IASCC), cracking due to cyclic loading (including flow-induced vibration), loss of material, loss of fracture toughness due to neutron or thermal embrittlement, and loss of preload due to thermal or irradiation-enhanced stress relaxation.

The program performs inspections for cracking and loss of material in accordance with the guidelines of applicable staff-approved BWRVIP documents and the requirements of ASME Code, Section XI, Table IWB-2500-1. The impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components. This program also manages loss of preload for core plate rim hold-down bolts and jet pump assembly hold-down beam bolts by performing visual inspections or stress analyses for adequate structural integrity. Enhanced guidance will be used for inspections of the Westinghouse (Nordic style) steam dryers.

Evaluations of reactor vessel internal component determined that supplemental inspections in addition to the existing BWRVIP examination guidelines are not necessary to manage loss of fracture toughness due to thermal aging

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embrittlement or neutron irradiation embrittlement and cracking due to IASCC during the second period of extended operation. This determination is based on neutron fluence, cracking susceptibility, fracture toughness, and consequences of cracking or failure of the reactor vessel internal components.

The BWR Vessel Internals aging management program will be enhanced to:

1. No longer applicable due to regulatory changes by revision of WRVIP-25 Rev 1-A and as analyzed in Engineering change 635331, "TECHNICAL EVALUATION TO JUSTIFY CORE PLATE BOLT INSPECTION ELIMINATION AT PEACH BOTTOM."
2. Perform a VT-3 inspection of the jet pump inlet mixer and beam regions every refuel cycle after a fluence value of $1.3E+20$ n/cm² (51 EFPY for Unit 2 and 63 EFPY for Unit 3) is reached at the jet pump holddown beam.
3. Perform periodic visual inspections of the PBAPS Westinghouse (Nordic style) stainless steel steam dryers for the aging effects of loss of material and cracking at a frequency not exceeding 10 years, with the first inspections performed prior to the second period of extended operation, as described below.

The inspection guidance contained in BWRVIP-139-A does not address the Westinghouse (Nordic style) steam dryers installed in PBAPS Unit 2 and Unit 3 and therefore is not directly applicable. However, the general principles and conclusions from BWRVIP-139-A, "BWR Vessel and Internals Project: Steam Dryer Inspection and Flaw Evaluation Guidelines", BWRVIP-181-R1-A, "BWR Vessel and Internals Project: Steam Dryer Repair Design Criteria", and Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing" were applied to the inspection plan described in WCAP-17635-P, "Peach Bottom Atomic Power Station Unit 2 and Unit 3 Replacement Steam Dryer Comprehensive Vibration Assessment Program (CVAP)". WCAP-17635-P also includes manufacturer's recommendations based on relevant operating experience. The scope of the inspection will include the items listed in Table 1 below.

The steam dryer inspections are based on the BWRVIP-139-A and WCAP-17635-P guidelines to identify loss of material (wear) and cracking using appropriate visual examination techniques (e.g., VT-1, VT-3) and qualified inspectors. The examination procedures identify the type and location of examination required for each dryer component as well as the reason for inspection. Acceptance criteria are consistent with BWRVIP-139-A and are described in procedures and work instructions. Flaws and abnormal indications identified will be entered into the corrective action program for engineering evaluation. The evaluations will consider increasing inspection frequency and scope as appropriate. Identified degradation left in the as found condition will be reinspected as required by the engineering evaluation.

The repair design criteria contained in BWRVIP-181-R1-A and BWRVIP-139-A will be used for any future repairs of the steam dryers. Repairs to the steam dryer will be inspected as specified in the repair design package.

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Table 1

Steam Dryer Inspection Program for the Second Period of Extended Operation in accordance with WCAP-17635-P

Inspection Location	Basis for Selection
1. Overall General Inspection of Outside of the Replacement Steam Dryer (to include outside of skirt)	Industry Operating Experience (BWRVIP-139-A, Section 1.1, Section 2.4.2) General Inspection for evidence of damage
2. Lifting Rods Top Ends (Unit 3 only)	Surfaces in contact during operation RG 1.20 Sec 2.3 (1)(b,d)
3. Hold Down Rods Top Ends (Unit 2 only)	Surfaces in contact during operation RG 1.20 Sec 2.3 (1)(b,d)
4. Support Ring Bottom Surface	RG 1.20 Sec 2.3 (1)(b,d) Surfaces in contact during operation
5. Outer hood (welds on outer surface)	Industry Operating Experience (BWRVIP-139-A, Section 1.1, Section 2.4.2) Higher stressed area identified in analysis (RG 1.20 Sec 2.3 (1)(e)) Inspection for evidence of IGSCC (weld not solution annealed)
6. Outer Ring Top Cage	Higher stressed area identified in analysis (RG 1.20 Sec 2.3 (1)(e)) Inspection for evidence of IGSCC (weld not solution annealed)
7. Weld attachments between the brackets to the lifting rod and hold down rod and weld attachments between the brackets to top plate (lifting rod, Unit 3 only)	Industry Operating Experience (BWRVIP-139-A, Section 2.4.8) Higher stressed area identified in analysis (RG 1.20 Sec 2.3 (1)(e)) Inspection for evidence of IGSCC (weld not solution annealed)

These enhancements will be implemented in accordance with the schedule described within each enhancement. Initial steam dryer inspections will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.8 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (3953023-08)

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) aging management program is a new condition monitoring program that will provide assurance that reactor coolant pressure boundary CASS components (i.e., pump casings) with the potential for significant thermal aging embrittlement meet their intended functions. The ASME Code Class 1 CASS components are maintained by inspecting and evaluating the extent of thermal aging embrittlement in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI. The PBAPS ASME Section XI

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Inservice Inspection program is augmented by the implementation of the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) program which will monitor the aging effect of loss of fracture toughness due to thermal aging embrittlement of ASME Code Class 1 CASS components with service conditions above 250 degrees Celsius (482 degrees Fahrenheit). PBAPS Unit 2 and Unit 3 do not have any Class 1 piping or fittings fabricated from CASS. The Class 1 reactor recirculation pump casings and covers are fabricated from CASS.

The program will include a screening methodology to determine components for which thermal aging embrittlement is potentially significant based on casting method, molybdenum content, and percent ferrite. Components with the potential for significant thermal aging embrittlement will be managed through either, qualified visual inspections, such as enhanced visual examination, or qualified ultrasonic testing methodology.

Inspections or evaluations are not required for components for which thermal aging embrittlement is not significant. In addition, screening for ASME Code Class 1 CASS valve bodies for significance of thermal aging embrittlement is not required, because the existing ASME Section XI inspection requirements are adequate for managing the aging effects of Class 1 valve bodies.

Reactor vessel internal components fabricated from CASS are not within the scope of this aging management program and are managed by the BWR Vessel Internals (R.2.1.7) aging management program.

This new aging management program will be implemented no later than six months prior to the second period of extended operation.

R.2.1.9 Flow-Accelerated Corrosion (3953023-09)

The Flow-Accelerated Corrosion aging management program is an existing condition monitoring program that manages wall thinning caused by flow-accelerated corrosion (FAC). The program is based on commitments made in response to NRC Generic Letter 89-08, "Erosion/Corrosion Induced Pipe Wall Thinning," and relies on implementation of the Electric Power Research Institute (EPRI) guidelines in the Nuclear Safety Analysis Center (NSAC)-202L-R4 for an effective FAC program.

CHECWORKS is used to predict component wear rates and remaining service life in the systems susceptible to FAC which provides reasonable assurance that structural integrity will be maintained between inspections. The model is revised if any changes in operating conditions or other factors that affect FAC (e.g., plant chemistry, power uprate) have occurred since the CHECWORKS model was last updated. Changes may also result from plant modifications that effect FAC behavior such as material changes, the addition of piping systems, piping system configuration changes, and the addition or replacement of in-line components. The CHECWORKS model is also refined by importing actual volumetric inspection data thickness measurements and re-running wear rate analyses. This improves the predictive capability of the model to ensure that intended functions are maintained.

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The program also manages wall thinning caused by mechanisms other than FAC in situations where periodic monitoring is used in lieu of eliminating the cause of various erosion mechanisms.

The program includes: (a) identifying all susceptible piping systems and components; (b) developing FAC predictive models to reflect component geometries, materials, and operating parameters; (c) performing analysis of FAC models and, with consideration of operating experience, selecting a sample of components for inspections; (d) inspecting components; (e) evaluating inspection data against acceptance criteria to determine the need for corrective actions including inspection sample expansion, repairs, or replacements, and to schedule future inspections; and (f) incorporating inspection data to refine FAC models.

The Flow-Accelerated Corrosion aging management program will be enhanced to:

1. Reassess infrequently used piping systems excluded from the scope of the program to ensure adequate bases exist to justify this exclusion for the second period of extended operation.

This enhancement will be implemented no later than six months prior to the second period of extended operation.

R.2.1.10 Bolting Integrity (3953023-10)

The Bolting Integrity aging management program is an existing condition monitoring program. The program manages aging for loss of preload, cracking, and loss of material of safety-related and nonsafety-related closure bolting on pressure-retaining components. The program utilizes recommendations and guidelines delineated in NUREG-1339, EPRI NP-5769, TR-1015336, and TR-1015337 for material selection, use of approved lubricants, proper torqueing, and leakage evaluations which are implemented during plant surveillance and maintenance activities.

In addition, the program manages aging of submerged mechanical bolting on the 2AS008, 2BS008, 3AS008, and 3BS008 Circulating Water Pump Structure intake traveling screens.

The program includes periodic visual inspections of closure bolting on pressure-retaining components for indication of loss of preload, cracking, and loss of material as evidenced by pressure-retaining joint leakage. Closure bolting on pressure-retaining components and mechanical bolting that are submerged or closure bolting on pressure-retaining components located in piping systems that contain air or gas is inspected by alternative means, such as by sample based periodic inspections. The program also includes preventive measures provided in the EPRI guidance documents to preclude or minimize loss of preload and cracking. Engineering procedures will be enhanced to clarify that recommended guidance for selection, storage and installation of bolting per applicable EPRI and Research Council on Structural Connections (RCSC) publications are a requirement.

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There is no high strength bolting material with actual yield strength of 150 ksi or greater on pressure-retaining components with bolt diameters greater than 2 inches, or bolts with unknown yield strength within the scope of this program. Therefore, sampling based volumetric examinations of closure bolting to detect indications of cracking is not applicable. Engineering procedures will be enhanced to require volumetric examination in accordance with ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, regardless of the code classification of the bolting, should high strength bolting greater than 2 inches in diameter be installed.

The program performs periodic sample based inspections on submerged closure bolting on the ESW, HPSW, and fire protection pumps; submerged closure bolting on the Core Spray, HPCI, RHR, and RCIC suction strainers; and submerged mechanical bolts on the 2AS008, 2BS008, 3AS008, and 3BS008 Circulating Water Pump Structure intake traveling screens. The program also performs periodic inspections on submerged closure bolting on the emergency cooling water pump.

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (R.2.1.1) program includes inspection of safety-related closure bolting on pressure-retaining components, and supplements this program. Inspection activities for bolting in a buried environment or underground with restricted access are performed in conjunction with buried piping and component inspections performed as part of the Buried and Underground Piping and Tanks (R.2.1.28) program.

The Reactor Head Closure Stud Bolting (R.2.1.3) program manages the aging effects of the bolting components for the reactor vessel closure head. The ASME Section XI, Subsection IWE (R.2.1.30) program, ASME Section XI, Subsection IWF (R.2.1.31) program; Structures Monitoring (R.2.1.34) program; RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (R.2.1.35) program; Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (R.2.1.13) program; manage the aging effects of safety-related and nonsafety-related structural bolting. The External Surfaces Monitoring of Mechanical Components (R.2.1.24) program manages the aging effects of safety-related and nonsafety-related bolting associated with ductwork for heating, ventilation, and air conditioning systems.

The Bolting Integrity aging management program will be enhanced to:

1. Ensure that submerged carbon steel closure bolts on the ESW, HPSW, and fire protection pumps are inspected for loss of material and to confirm that the closure bolting is hand tight. A minimum of 19 bolt inspections shall be performed each 10-year period during the second period of extended operation for each unit. Inspection of closure bolting on these pumps during pump overhaul and replacement activities may be credited during each 10-year period in the second period of extended operation.
2. Ensure that submerged stainless steel mechanical bolts on the 2AS008, 2BS008, 3AS008, and 3BS008 Circulating Water Pump Structure intake traveling screens are inspected for loss of material and to confirm that the

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mechanical bolting is hand tight. A minimum of 19 bolt inspections shall be performed each 10-year period during the second period of extended operation for each unit. Inspection of mechanical bolting on these screens during overhaul and replacement activities may be credited during each 10-year period in the second period of extended operation.

3. Ensure that closure bolts on pressure-retaining components that contain air or gas are inspected for cracking and loss of material for the carbon steel/ air-indoor uncontrolled and the stainless steel/ air-indoor uncontrolled material and environment combinations. In addition, the inspections will confirm that this closure bolting is leak tight applying inspection techniques, such as soap bubble testing, thermography, acoustic testing, or verifying closure bolting is hand tight. A minimum of 19 bolt inspections shall be performed each 10-year period during the second period of extended operation for each unit. Opportunistic inspections during maintenance activities may be credited during the same 10-year period.

4. Ensure that closure bolts on pressure-retaining components that contain air or gas are inspected for loss of material for the carbon steel/ air-outdoor material and environment combination. In addition, the inspections will confirm that this closure bolting is leak tight applying inspection techniques, such as soap bubble testing, thermography, acoustic testing, or verifying closure bolting is hand tight. A minimum of 25 bolt inspections shall be performed each 10-year period during the second period of extended operation for both Units 2 and 3. Opportunistic inspections during maintenance activities may be credited during the same 10-year period.

5. Revise site walkdown procedures to specify proper lighting and appropriate distances to adequately identify visible component leakage, evidence of past leakage, or other age-related degradation on pressure-retaining bolted joints that contain fluids such as water, oil, or steam. Cameras and video equipment may be used to supplement these inspections.

6. Revise existing repetitive tasks to provide guidance for proper lighting and appropriate inspection distances to adequately identify loss of material in submerged environments. Cameras and video equipment may be used to supplement these inspections.

7. Ensure no fewer than five additional bolts are inspected for each sample based inspection that does not meet acceptance criteria, or 20 percent of the total bolt population of each applicable material, environment, and aging effect combination; whichever is less. If these subsequent inspections do not meet acceptance criteria, an extent of condition and extent of cause analysis are performed to determine the further extent of inspections. These additional inspections will be completed within the inspection interval for which the original sample based inspections are conducted.

8. Revise engineering procedures to require volumetric examination in accordance with ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, regardless of the code classification of the bolting, should high strength bolting greater than 2 inches in diameter be installed.

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9. Clarify that the recommended guidance for proper selection of bolting material and lubricants, and appropriate installation torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting is a requirement at Peach Bottom in accordance with the guidelines provided in EPRI NP-5067 and TR-104213. Clarify that the recommended requirements for storage, lubricant selection, and bolting and coating material selection to include the recommendations in Section 2 of Research Council on Structural Connections (RCSC) publication "Specification for Structural Joints Using High Strength Bolts," are a requirement at Peach Bottom.

These enhancements will be implemented no later than six months prior to the second period of extended operation.

R.2.1.11 Open-Cycle Cooling Water System (3953023-11)

The Open-Cycle Cooling Water System aging management program is an existing preventive, mitigative, condition monitoring, and performance monitoring program based on the implementation of NRC GL 89-13, and includes nonsafety-related portions of the open cycle cooling water system. The program includes: (a) surveillance and control to significantly reduce the incidence of flow blockage problems as a result of biofouling, (b) tests to verify heat transfer of heat exchangers, (c) periodic inspection and maintenance so that corrosion, erosion, cracking, fouling, and biofouling cannot degrade the performance of systems serviced by the open cycle cooling water system. This program includes guidance beyond the requirements contained in NRC GL 89-13, such as inputs from industry reports and documents (e.g., EPRI documents) that address operating experience such that aging effects are adequately managed.

The Open-Cycle Cooling Water System aging management program will be enhanced to:

1. Provide procedural direction to perform additional inspections if the cause of the aging effect for each applicable material and environment combination is not corrected by repair or replacement for all components constructed of the same material and exposed to the same environment. These additional inspections will be conducted if any of the inspections do not meet acceptance criteria. No fewer than five additional inspections will be performed for each inspection that does not meet acceptance criteria, or 20 percent of each applicable material, environment, and aging effect combination, whichever is less.

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2. Perform a minimum of 20 inspections for recurring internal corrosion in the raw water cooling water systems every 24 months until the rate of recurring internal corrosion occurrences no longer meets the criteria for recurring internal corrosion as defined in SLRA Section 3.3.2.2.7. The selected inspection locations will be periodically reviewed to validate their relevance and usefulness and adjusted as appropriate. Evaluation of the inspection results will include (1) a comparison to the nominal wall thickness or previous wall thickness measurements to determine rate of corrosion degradation; (2) a comparison to the design minimum allowable wall thickness to determine the acceptability of the component for continued use; and (3) a determination of reinspection interval.

3. Provide procedural direction to require the use of a mill tolerance of 12.5% for added conservatism when determining corrosion rates at new inspection locations if corrosion rates from other locations with nearly identical operating conditions, material, size, and configuration cannot be used.

These enhancements will be implemented no later than six months prior to the second period of extended operation.

R.2.1.12 Closed Treated Water Systems (3953023-12)

The Closed Treated Water Systems program is an existing mitigative program that manages loss of material, cracking, and reduction of heat transfer in piping, piping components, tanks, and heat exchangers exposed to a closed cycle cooling water environment. The program will be enhanced to include condition monitoring activities. The program includes: (a) nitrite-based water treatment, including pH control and the use of corrosion inhibitors, to modify the chemical composition of the water such that the function of the equipment is maintained and such that the effects of corrosion are minimized; (b) chemical testing of the water to ensure that the water treatment program maintains the water chemistry within acceptable guidelines; and (c) inspections to determine the presence or extent of corrosion, stress corrosion cracking, or fouling.

The program uses EPRI guidelines for chemistry control of closed cooling water systems. Corrosion coupon testing is not used; corrosion monitoring is performed by monitoring for total iron and total copper which indicates if active corrosion is occurring. Testing and treating for microbiological growth is performed.

Periodic inspections of a representative sample will be performed, at a minimum, in each 10-year period during the second period of extended operation. The inspections will include opportunistic visual inspections and periodic inspections to satisfy the sample size requirements using techniques capable of detecting loss of material, cracking, and fouling, as appropriate to verify the effectiveness of water chemistry control to mitigate aging effects. Opportunistic visual inspections will be performed whenever a system boundary is opened.

The inspections will focus on the components most susceptible to aging because of time in service and severity of operating conditions, including locations where local conditions may be significantly more severe than those in

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the bulk water. A representative sample is 20 percent of the population (defined as components having the same material, water treatment program, and aging effect combination) or a maximum of 19 components per population at each unit. At least 20 percent of the surface area will be inspected unless the component is measured in linear feet, such as piping. In that case, any combination of 1-foot length sections and components can be used to meet the recommended extent of 19 inspections per unit. Additional inspections will be conducted if one of the inspections does not meet acceptance criteria. The number of increased inspections will be determined in accordance with the corrective action program; however, no fewer than five additional inspections for each inspection that did not meet acceptance criteria, or 20 percent of each applicable material, environment, and aging effect combination is inspected, whichever is less. If subsequent inspections do not meet acceptance criteria, an extent of condition and extent of cause analysis will be conducted to determine the further extent of inspections. Additional components will be inspected for any recurring degradation to ensure corrective actions appropriately address the associated causes. The additional inspections will include inspections at both Unit 2 and 3 with the same material, environment, and aging effect combination. The additional inspections will be completed within the interval in which the original inspection was conducted.

Due to the water chemistry controls, no age-related degradation is expected; therefore, any detectable loss of material, cracking, or fouling will be evaluated in the corrective action program. Identified age-related degradation will be projected until the next scheduled inspection and results evaluated to confirm that the sampling bases will maintain the component's intended functions throughout the second period of extended operation. If fouling is identified, the overall effect will be evaluated for reduction of heat transfer, flow blockage, and loss of material.

Inspections will be performed by personnel qualified in accordance with site procedures and programs to perform the specified task. Inspections within the scope of the ASME Code will follow procedures consistent with the ASME Code. Non-ASME Code inspections will follow site procedures that include requirements for items such as lighting, distance, offset, surface coverage, presence of protective coatings, and cleaning processes.

The Closed Treated Water Systems aging management program will be enhanced to:

1. Perform condition monitoring including opportunistic visual inspections and sample-based periodic inspections using techniques (visual, surface, or volumetric) capable of detecting loss of material, cracking, and fouling, as appropriate to verify the effectiveness of water chemistry control to mitigate aging effects in each 10-year period during the second period of extended operation. The rate of identified degradation will be projected until the next scheduled inspection. Additional sample-based inspections will be performed if aging effects are identified. If those inspections identify aging effects, the corrective action program will be used to determine the extent of condition and extent of cause to determine the further extent of inspections.

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This enhancement will be implemented no later than six months prior to the second period of extended operation.

R.2.1.13 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (3953023-13)

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems aging management program is an existing condition monitoring program that manages the effects of loss of material due to corrosion and wear, cracking, deformation, and indications of loss of preload for load handling bridges, structural members, structural components, and bolted connections. Procedures and controls implement the guidance on the control of overhead heavy load cranes specified in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The program utilizes periodic visual inspections as described in the ASME B30 series of standards for inspection, detection of aging effects, evaluation, and repair of aging effects. Monitoring and maintenance of structural components of crane handling systems follow the maintenance rule requirements provided in 10 CFR 50.65.

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems aging management program will be enhanced to:

1. Provide additional guidance to include inspection of crane-related bridges, structural members, and structural components for deformation, cracking, and loss of material due to corrosion or wear; and associated bolted connections for loss of material, cracking, and indications of loss of preload.
2. Provide procedural direction to document deficiencies identified during inspection activities within the corrective action program.
3. Provide site-specific procedural direction to evaluate and repair visual indication of loss of material, deformation, or cracking, and any visual sign of loss of bolting preload in accordance with ASME B30.2 or other applicable industry standard in the ASME B30 series.

These enhancements will be implemented no later than six months prior to the second period of extended operation.

R.2.1.14 Compressed Air Monitoring (3953023-14)

The Compressed Air Monitoring aging management program is an existing condition monitoring program that consists of monitoring moisture content and corrosion, and performance of the compressed air system, including: (a) preventive monitoring of water (moisture), and other contaminants to keep within the specified limits, and (b) inspection of components for indications of loss of material due to corrosion.

This program is based on the PBAPS response to NRC GL 88-14 and INPO's SOER 88-01. It also relies on the guidance from the ASME operations and maintenance standards and guides (ASME OM-S/G-2012, Division 2, Part 28) and ANSI/ISA-7.0.1-1996, and EPRI TR-10847 for testing and monitoring air quality and moisture. Additionally, periodic opportunistic visual inspections of

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component internal surfaces will be performed for signs of loss of material due to corrosion. Program activities include air quality checks at various locations to ensure the dew point, particulates, and hydrocarbons are maintained within specified limits, and inspections of the internal surfaces of compressed air system components for signs of loss of material due to corrosion.

The Compressed Air Monitoring aging management program will be enhanced to:

1. Perform daily inspection of instrument nitrogen after dryer desiccant for signs of moisture. Results will be recorded and reviewed to determine if corrective actions are required.
2. Perform opportunistic visual inspections of component internal surfaces exposed to a dry air environment for signs of loss of material due to corrosion.

These enhancements will be implemented no later than six months prior to the second period of extended operation.

R.2.1.15 BWR Reactor Water Cleanup System (3953023-15)

The BWR Reactor Water Cleanup System program is an existing condition monitoring and mitigation program that includes the requirements to perform augmented inservice inspection (ISI) to manage stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) for stainless steel Reactor Water Cleanup (RWCU) system piping welds outboard of the second (outboard) primary containment isolation valves. Piping that contains reactor coolant at a temperature above 200 degrees Fahrenheit during power operation, and has a nominal diameter of 4 inches or larger, regardless of ASME Code classification is included within the scope of the program. The program includes the measures delineated in NUREG-0313, Revision 2, NRC Generic Letter (GL) 88-01 and its Supplement 1. For PBAPS Units 2 and 3, the NRC previously approved a reduced extent and schedule for inspection of IGSCC susceptible welds within the RWCU system that are outboard of the second isolation valves. The approved alternate extent and schedule is to perform ISI on two percent of the IGSCC susceptible welds during each refueling outage. The program is implemented in conjunction with the Water Chemistry (R.2.1.2) program to minimize the potential of cracking due to SCC or IGSCC in a treated water environment.

R.2.1.16 Fire Protection (3953023-16)

The Fire Protection aging management program is an existing condition monitoring and performance monitoring program that includes fire barrier visual inspections and low pressure carbon dioxide system visual inspections and functional testing. The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, floors, fire dampers, and other materials that perform a fire barrier function. Periodic visual inspection and functional testing of the fire rated doors is performed to ensure that their functionality is maintained and aging effects managed. The program also includes visual inspections and periodic functional tests of the low pressure carbon dioxide fire suppression systems using the National Fire

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Protection Association Codes and Standards for guidance.

The Fire Protection aging management program will be enhanced to:

1. Perform periodic visual inspection every 18 36 months for identification of corrosion that may lead to loss of material on the external surfaces of the low pressure carbon dioxide fire suppression systems.
2. Perform periodic visual inspection of combustible liquid spill retaining curbs every 24 months.

These enhancements will be implemented no later than six months prior to the second period of extended operation.

R.2.1.17 Fire Water System (3953023-17)

The Fire Water System aging management program is an existing condition monitoring program that manages aging effects such as loss of material and flow blockage due to fouling associated with water-based fire protection system components. The program manages these aging effects through the use of system pressure monitoring, system header flushing, underground main supply flow testing, pump performance testing, hydrant flushing, sprinkler system and deluge system flow testing, visual inspections, and volumetric examinations performed using the guidance of NFPA 25, 2011 Edition.

The program does not manage cracking as an aging effect because the PBAPS Fire Water System does not contain fire water storage tanks or high density polyethylene (HDPE) pipe. The fire main underground cement lined pipe aging effects including cracking are managed by the Internal Coating/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (R.2.1.29) program.

Testing or replacement of sprinklers that have been in place for 50 years is performed using the guidance of NFPA 25, 2011 Edition.

PBAPS does not have water-based fire protection systems that are normally dry but periodically subject to flow and cannot be drained. Therefore, augmented testing in addition to that specified in NFPA 25 is not required.

The water-based fire protection system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated.

Visual inspections and volumetric wall thickness examinations are used to detect corrosion and loss of material. The visual inspection techniques are capable of detecting surface irregularities that could indicate an unexpected level of degradation due to corrosion and corrosion product deposition. When visual inspections detect such irregularities, follow-up volumetric wall thickness examinations are performed. When the presence of foreign material that could cause flow blockage is found, the material will be removed and the source corrected by the use of the corrective action program. Inspections and tests follow site procedures that include specific inspection guidance and parameters

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such as lighting, distance, offset, protective coatings, and cleaning processes for an adequate examination.

The Fire Water System aging management program will be enhanced to:

1. Revise flow test procedures to include:
 - a. Inspector test flush acceptance criteria for wet pipe sprinkler systems that currently do not include the requirement to record time to flow from the opened test valve.
 - b. Acceptance criteria for wet pipe main drain tests. Flowing pressures from test to test will be monitored to determine if there is a 10 percent reduction in full flow pressure when compared to previously performed tests. An issue report shall be generated in the corrective action program to determine the cause and corrective actions.
 - c. If flow test acceptance criteria are not met, perform an investigation within the corrective action program that includes review for increased testing and perform at least two successful additional tests. Additional tests shall be completed within the interval in which the original test was conducted. If acceptance criteria are not met during follow-up testing, an extent of condition and extent of cause analysis shall be conducted to determine the further extent of tests which includes testing on the same system, on the other unit.
2. Perform air flow tests on the hydrogen seal oil and reactor building water curtains every two years to ensure deluge piping and nozzles are unobstructed and there are no flow blockages.
3. Increase the frequency of air flow tests through the standby gas treatment and recombiner system deluge piping and nozzles to every two years to ensure piping and nozzles are unobstructed and there are no flow blockages.
4. Revise procedures to improve guidance for external visual inspections of the in scope sprinkler systems piping and sprinklers at least every two years to inspect for corrosion, loss of material, leaks, and proper sprinkler orientation. Corroded, leaking or damaged sprinklers shall be replaced.
5. Perform external visual inspections of the in scope above ground fire main piping every two years to identify excessive corrosion, loss of material, leaks, and physical damage.
6. Perform internal visual inspections of sprinkler and deluge system piping to identify internal corrosion, foreign material, and obstructions to flow. Follow-up volumetric wall thickness examinations will be performed if internal visual inspections detect an unexpected level of degradation due to corrosion and corrosion product deposition. If organic or foreign material, or internal flow blockage that could result in failure of system function is identified, then an obstruction investigation will be performed within the corrective action program that includes removal of the material, an extent of condition determination, review for increased inspections, extent of follow-up examinations, and a flush

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in accordance with NFPA 25 Annex D.5, Flushing Procedures. The internal visual inspections will consist of the following:

- a. Wet pipe sprinkler systems - 50 percent of the wet pipe sprinkler systems in scope for license renewal will have visual internal inspections of piping by removing a hydraulically remote sprinkler, performed every five years, consistent with NFPA 25, 2011 Edition, Section 14.2. During the next five-year inspection period, the alternate systems previously not inspected shall be inspected.
 - b. Pre-action sprinkler systems - pre-action sprinkler systems in scope for license renewal will have visual internal inspections of piping by removing a hydraulically remote nozzle, performed every five years, consistent with NFPA 25, 2011 Edition, Section 14.2.
 - c. Deluge systems - Yard transformer deluge systems in scope for license renewal will have visual internal inspections of piping by removing a hydraulically remote nozzle, performed every five years, consistent with NFPA 25, 2011 Edition, Section 14.2.
7. Perform a one-time volumetric wall thickness inspection, prior to the second period of extended operation, on a sample of the original yard transformer deluge system supply piping that was not replaced during transformer replacements and is periodically subjected to flow during functional testing.
 8. Revise service water bay inspection procedures to include inspection of the motor driven fire pump intake strainer.
 9. Perform flow tests for hose stations at the hydraulically most limiting locations for each zone of the system on a five-year frequency to demonstrate the capability to provide the design pressure at required flow.
 10. Flush deluge system mainline supply basket strainers until clear, following functional testing of yard deluge systems.
 11. Perform a one-time inspection of the auxiliary boiler fuel oil storage tank internal foam nozzle and deflector, prior to the second period of extended operation, to ensure proper configuration and orientation and no indication of flow blockage.
 12. Perform an internal inspection of the auxiliary boiler oil storage tank foam system foam concentrate tank every 10 years to ensure it is free of corrosion, debris, or foreign material that could cause flow blockage, and to ensure there are no cracks or leaks and it is in good condition.
 13. Revise restoration procedures for the hydrogen seal oil and reactor building water curtain systems to utilize low point drains following control valve actuations to ensure there is no trapped water in the system.
 14. Revise restoration procedures for the yard transformer deluge systems to utilize low point drains after functional testing.

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15. Revise the fire hydrant inspection and flush test procedure to include a minimum flow duration of one (1) minute after the hydrant valve is fully open to remove all foreign material.

16. Revise the underground fire main flow test to utilize the corrective action program to determine an increased test frequency when established test criteria is not met or when significant degraded trends that could adversely affect system intended function are identified. When test results pass the established test criteria, the test frequency may be extended to a five (5) year frequency IAW NFPA 25.

17. Perform at least five additional ultrasonic test inspections on the fire water supply piping for each Fire Water System pipe wall inspection that does not meet acceptance criteria.

18. Provide procedural direction to require the use of a mill tolerance of 12.5% for added conservatism when determining corrosion rates at new inspection locations if corrosion rates from other locations with nearly identical operating conditions, material, size, and configuration cannot be used.

These enhancements will be implemented no later than six months prior to the second period of extended operation. Inspections that are to be completed prior to the second period of extended operation will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.18 Outdoor and Large Atmospheric Metallic Storage Tanks (3953023-18)

The Outdoor and Large Atmospheric Metallic Storage Tanks aging management program is an existing condition monitoring program that manages aging effects associated with in-scope outdoor aboveground tanks constructed on concrete or soil. PBAPS has no indoor, large volume tanks containing water designed with internal pressures approximating atmospheric pressure that are sited on concrete or soil, and no indoor tanks that sit on, or are embedded in concrete, where specific operating experience indicates that the tank surfaces are periodically exposed to moisture. The scope of this program includes the Unit 2 condensate storage tank, the Unit 3 condensate storage tank, and the common refueling water storage tank. These tanks contain treated water, are constructed of carbon steel, are not insulated, are coated both internally and externally as a preventive measure to mitigate corrosion, and are supported on a concrete or asphalt and sand foundation, such that the bottoms of the tanks are inaccessible for direct visual inspection. Sealant is used at the concrete or asphalt interface with the tank.

The program manages loss of material by conducting periodic internal and external visual inspections on a frequency of 10 years or less. Cracking is not a predicted aging effect due to the carbon steel construction. Visual inspections of sealant and caulking will be supplemented with physical manipulation to detect degradation. Thickness measurements of tank bottoms are conducted to ensure that significant degradation is not occurring. Inspections are conducted in accordance with plant-specific procedures

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including inspection parameters such as lighting, distance, offset, and surface conditions, as defined for visual and ultrasonic inspection techniques.

The Outdoor and Large Atmospheric Metallic Storage Tanks aging management program will be enhanced to:

1. Perform a visual inspection of the sealant at the perimeter of the condensate storage tanks and refueling water storage tank bases for signs of degradation every two years. The visual inspections of sealant and caulking are supplemented with physical manipulation to detect degradation.
2. Perform a pre-inspection review of the previous two inspections of the internal tank coatings, when available, that includes review of results of inspections and any subsequent repair activities.
3. Conduct training and qualification of individuals involved in internal coating or lining inspections and evaluating degraded conditions in accordance with an ASTM International standard endorsed in RG 1.54.
4. Perform volumetric inspection of Unit 2 and 3 condensate storage tanks and refueling water storage tank bottoms at least once during the 10-year period prior to the second period of extended operation, and at least once every 10 years during the second period of extended operation. Volumetric inspections are performed at representative sample locations to include 25 one square foot locations or 20 percent coverage conducted in different locations unless the program states the basis for why repeated inspections are conducted in the same location (i.e. previous findings). Additionally, a minimum of 10 of the random one square foot sample locations will be performed within the 30-inch band at the perimeter of the shell. The scope of subsequent examinations may be adjusted based upon the results of previous examinations.

These enhancements will be implemented no later than six months prior to the second period of extended operation, unless a more specific schedule is described within the enhancement (i.e., Enhancement 4). Inspections that are to be completed prior to the second period of extended operation will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.19 Fuel Oil Chemistry (3953023-9)

The Fuel Oil Chemistry aging management program is an existing mitigative and condition monitoring program that includes activities which provide assurance that contaminants are maintained at acceptable levels in fuel oil for systems and components within the scope of license renewal. The program relies on a combination of surveillance and maintenance procedures. The fuel oil tanks within the scope of license renewal are maintained by monitoring and controlling fuel oil contaminants in accordance with the Technical Specifications, Technical Requirements Manual, and ASTM guidelines. Exposure to fuel oil contaminants, such as water and microbiological organisms, is minimized by periodic cleaning and draining of tanks and by verifying the quality of fuel oil. Fuel oil sampling and analysis is performed in

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accordance with approved procedures for stored fuel oil and new fuel oil before its introduction into the storage tanks.

The Fuel Oil Chemistry aging management program will be enhanced to:

1. Perform periodic internal inspection of the diesel fire pump fuel oil storage tank (00T041) and the diesel fire pump day tank (00T543) at least once during the 10-year period prior to the second period of extended operation, and at least once every 10 years during the second period of extended operation. Each diesel fuel tank will be drained and cleaned, the internal surfaces visually inspected (if physically possible), and, if evidence of degradation is observed during inspections, or if visual inspection is not possible, these diesel fuel tanks will be volumetrically inspected.
2. Perform periodic (quarterly) removal of water collected at the bottom of the diesel fire pump fuel oil storage tank (00T041) and the diesel fire pump day tank (00T543).
3. Perform receipt testing of new fuel oil for particulate concentration and the levels of microbiological organisms for the diesel generator fuel oil day tanks (0A(B,C,D)T040), diesel generator fuel oil storage tanks (0A(B,C,D)T038), and diesel fire pump fuel oil storage tank (00T041).
4. Perform periodic (quarterly) sampling and analysis for water and sediment content, particulate concentration, and the levels of microbiological organisms for the diesel generator fuel oil day tanks (0A(B,C,D)T040). Sampling activities will include a sampling methodology that includes a representative sample from the lowest point in the tank.
5. Perform periodic (quarterly) sampling and analysis for water and sediment and the levels of microbiological organisms for the diesel generator fuel oil storage tanks (0A(B,C,D)T038).
6. Perform periodic (quarterly) sampling and analysis for particulate concentration and the levels of microbiological organisms for the diesel fire pump fuel oil storage tank (00T041) and the diesel fire pump day tank (00T543).
7. Perform periodic (quarterly) trending of water and sediment content, particulate concentration, and the levels of microbiological organisms for all fuel oil tanks within the scope of the program.
8. Evaluate the need for biocide or corrosion inhibitor addition if periodic testing indicates biological activity or evidence of corrosion.
9. Evaluate degradation identified during tank internal inspections against acceptance criteria to confirm that the timing of subsequent inspections will maintain the components' intended function throughout the second period of extended operation based on the projected rate of degradation.

These enhancements will be implemented no later than six months prior to the second period of extended operation unless a more specific schedule is

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described within the enhancement (i.e., Enhancement 1). Inspections that are to be completed prior to the second period of extended operation will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.20 Reactor Vessel Material Surveillance (3953023-20)

The Reactor Vessel Material Surveillance aging management program is an existing condition monitoring program that manages the loss of fracture toughness due to neutron irradiation embrittlement of the ferritic reactor pressure vessel (RPV) beltline materials in a reactor coolant and neutron flux environment. The program utilizes surveillance capsules that are located near the inside wall of the RPV beltline region in order to duplicate the neutron spectrum, temperature history, and neutron fluence of the RPV inner surface. The resulting lead factor allows the surveillance capsules to achieve a neutron fluence exposure earlier than the RPV allowing the surveillance capsules to be withdrawn and tested prior to the RPV reaching the neutron fluence of interest.

PBAPS, Unit 2 and Unit 3 are currently participating in an NRC approved integrated surveillance program (ISP) which covers the first period of extended operation. Unit 2 is a BWRVIP ISP host plant with three capsules in the vessel, one previously tested and reconstituted and two untested. The two untested capsules are scheduled to be withdrawn and tested during the first period of extended operation in accordance with the BWRVIP ISP. PBAPS Unit 3 is a BWRVIP ISP non-host plant with three capsules in the vessel, one previously tested and reconstituted and two untested. Unit 3 is not scheduled for withdrawal of any capsules during the first period of extended operation and relies on capsule test data from River Bend Nuclear Generating Station, Duane Arnold Energy Center, and the BWROG Supplemental Surveillance Program (SSP) in accordance with the BWRVIP ISP. The NRC has not approved an ISP for the second period of extended operation. For the second period of extended operation, the program will be enhanced to implement a reactor specific in-vessel program, complying with ASTM E 185-82, which will withdraw and test a surveillance capsule during the second period of extended operation, with a neutron fluence of the capsule between one and two times the projected peak vessel neutron fluence at the end of the second period of extended operation. For boiling water reactors, the peak neutron fluence of interest is the 1/4T peak neutron fluence at the end of the second period of extended operation.

The program provides sufficient material data and dosimetry to: (a) monitor irradiation embrittlement neutron fluences greater than the projected neutron fluence at the end of the second period of extended operation, and (b) provide adequate dosimetry monitoring during the operational period. A surveillance capsule will be withdrawn and tested from each reactor vessel during the second period of extended operation providing reactor vessel material irradiation embrittlement data and dosimetry monitoring during the second period of extended operation.

The program is a condition monitoring program that measures the increase in

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Charpy V-notch 30 foot-pound (ft-lb) transition temperature and the drop in upper-shelf energy as a function of neutron fluence and irradiation temperature. RPV beltline material test results provide reactor vessel material fracture toughness data for the neutron irradiation embrittlement time-limited aging analyses (TLAAs) (e.g., upper-shelf energy and pressure-temperature limits evaluations). The RPV beltline material surveillance capsules are removed at various exposure intervals for monitoring and trending purposes and in conjunction with the Neutron Fluence Monitoring (R.3.1.2) program. See Section R.4.2 for discussion of the TLAAs associated with neutron irradiation embrittlement.

Surveillance capsule are withdrawn, tested, and results reported in accordance with 10 CFR Part 50, Appendix H and ASTM E 185-82. Any changes to the surveillance capsule withdrawal schedule, including changing the status of standby capsules, must be approved by the NRC prior to implementation per 10 CFR Part 50, Appendix H. Specimens from tested capsules and withdrawn untested capsules are maintained in storage for possible reconstitution or in-insertion.

The Reactor Vessel Material Surveillance aging management program will be enhanced to:

1. Withdraw and test the Unit 2, 120 degree reconstituted capsule and the Unit 3, 120 degree capsule per the capsule withdrawal schedules below. A technical summary report containing the test results shall be submitted to the NRC per the requirements of 10 CFR Part 50, Appendix H. Any changes to the Reactor Vessel Material Surveillance program must be submitted for NRC review and approval in accordance with 10 CFR Part 50, Appendix H.

Peach Bottom Unit 2 Capsule Withdrawal Schedule		
Capsule	Capsule Lead Factor (OT/¼T)	Capsule Withdrawal EFPY
30°	0.95/1.38	Per BWRVIP-86-R1-A
120°	0.95/1.38	7.53 (actual)
120° Reconstituted	0.95/1.38	60 - 62 ⁽¹⁾
300°	0.95/1.38	Per BWRVIP-86-R1-A

1. Capsule 120° was withdrawn, tested, and reconstituted after Cycle 7 and re-inserted after Cycle 8, therefore capsule EFPY is 1.21 EFPY less than plant operating EFPY.

Peach Bottom Unit 3 Capsule Withdrawal Schedule		
Capsule	Capsule Lead Factor (OT/¼T)	Capsule Withdrawal EFPY
30°	0.95/1.38	7.57 (actual)
30°	0.95/1.38	Spare ⁽¹⁾

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Reconstituted		
120°	0.95/1.38	60 - 62
300°	0.95/1.38	Spare ⁽²⁾

1. Capsule 30° was withdrawn, tested, and reconstituted after Cycle 7 and re-inserted after Cycle 8, therefore capsule EFPY is 1.41 EFPY less than plant operating EFPY.
2. Capsule 300° was withdrawn after Cycle 7 and re-inserted after Cycle 8, therefore capsule EFPY is 1.41 EFPY less than plant operating EFPY.

This enhancement will be implemented in accordance with the schedules defined in the enhancement.

R.2.1.21 One-Time Inspection (3953023-21)

The new One-Time Inspection aging management program is a condition monitoring program consisting of a one-time inspection of selected components to verify: (a) the system-wide effectiveness of an AMP that is designed to prevent or minimize aging to the extent that it will not cause the loss of intended function during the second period of extended operation; (b) the insignificance of an aging effect; and (c) that long-term loss of material will not cause a loss of intended function for steel components exposed to environments that do not include corrosion inhibitors as a preventive action.

The elements of the program will include: (a) determination of the sample size of components to be inspected based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience, (b) identification of the inspection locations in the system or component based on the potential for the aging effect to occur, (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined, and (d) an evaluation of the need for follow-up examinations to monitor the progression of aging if age-related degradation is found that could jeopardize an intended function before the end of the second period of extended operation.

Periodic inspections instead of this program will be used for structures or components with known age-related degradation mechanisms or when the environment in the second period of extended operation is not expected to be equivalent to that in the prior operating period. Inspections not conducted in accordance with ASME Code Section XI requirements will be conducted in accordance with plant-specific procedures including inspection parameters such as lighting, distance, offset, and surface conditions.

This new aging management program will be implemented no later than 10 years prior to the second period of extended operation. The one-time inspections are required to be performed within the 10 years prior to the second period of extended operation, and no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

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R.2.1.22 Selective Leaching (3953023-22)

The Selective Leaching aging management program is a new condition monitoring program that will monitor components constructed of materials which are susceptible to selective leaching. Susceptible materials are gray cast iron, ductile iron, and copper alloys that contain greater than 15 percent zinc. Copper alloys containing greater than 8 percent aluminum (aluminum bronze) are also susceptible to selective leaching; however, there are no components in the scope of license renewal that are constructed of this material. The selective leaching program includes a one-time inspection for susceptible components exposed to closed cycle cooling water and treated water environment since plant-specific operating experience has not revealed selective leaching in these environments, as well as opportunistic and periodic inspections for susceptible components exposed to raw water, waste water, and soil (which may include groundwater) environments.

Visual inspections supplemented by mechanical examination techniques such as chipping or scraping (for ductile and gray cast iron components) will be conducted on a representative sample of susceptible components. In addition, periodic destructive examinations of components for physical properties (i.e., degree of dealloying, depth of dealloying, through wall thickness, and chemical composition) will be conducted for components exposed to raw water, waste water, and soil environments. Inspections and tests will be conducted to determine whether loss of material will affect the ability of the components to perform their intended function for the second period of extended operation. Inspections will be conducted in accordance with plant-specific procedures including inspection parameters such as lighting, distance, offset and surface conditions as appropriate. When the acceptance criteria are not met such that it is determined that the affected component should be replaced prior to the end of the second period of extended operation, additional inspections will be performed.

This new aging management program will be implemented no later than 10 years prior to the second period of extended operation. The one-time inspections are required to be performed within the 10 years prior to the second period of extended operation, and no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.23 ASME Code Class 1 Small-Bore Piping (3953023-23)

The ASME Code Class 1 Small-Bore Piping aging management program is a new condition monitoring program that augments the existing ASME Code, Section XI requirements and is applicable to ASME Code Class 1 small-bore piping and systems with a NPS diameter less than 4 inches and greater than or equal to 1 inch. This program provides for volumetric examination of a sample of full penetration (butt) welds and partial penetration (socket) welds in Class 1 piping to manage cracking due to stress corrosion cracking or thermal or vibratory fatigue loading. Volumetric examinations will employ techniques that have been demonstrated to be capable of detecting flaws and discontinuities in the examination volume of interest. Destructive examination methods may be

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used in lieu of volumetric examination. The program includes measures to verify that degradation is not occurring; thereby either to confirm that there is no need to further manage aging-related degradation or to validate the effectiveness of existing programs and practices for the second period of extended operation.

The extent and schedule for volumetric examination is based on plant-specific operating experience and whether actions have been implemented that effectively mitigate the cause(s) of any past cracking. The program provides for a one-time inspection of a sample of the population of welds (butt welds or socket welds) for plants that have not experienced cracking or have experienced cracking but have implemented corrective actions, such as a design change, to effectively mitigate the cause(s) of the cracking. The program provides for periodic inspection of a sample of the population of welds (butt welds or socket welds) that have experienced cracking and have not implemented corrective actions to effectively mitigate the cause(s) of the cracking.

The only two instances of cracking of Class 1 small-bore piping occurred at socket welds on Unit 3 in 2005 and in 2017. The causes of these cracking conditions were effectively mitigated by design changes. There have been no Class 1 small-bore piping cracking events at Unit 3 butt welds or at Unit 2 socket welds or butt welds. For butt welds, the one-time inspection sample size will be three percent up to a maximum of 10 inspections for Unit 2 and Unit 3. Since there are 89 Unit 2 butt welds and 90 Unit 3 butt welds, three butt welds on Unit 2 and Unit 3 will be inspected. For socket welds, the one-time inspection sample size will be three percent up to a maximum of 10 on Unit 2, and 10 percent up to a maximum of 25 on Unit 3. Since there are more than 1000 socket welds on each Unit, 10 socket welds on Unit 2 and 25 socket welds on Unit 3 will be inspected. If destructive examination is used, then each weld receiving a destructive examination can be credited as equivalent to two volumetric examinations.

If cracking is revealed by a one-time inspection, then additional one-time inspections will be performed for the population of welds (butt welds or socket welds) that have experienced cracking in accordance with ASME Section XI, Subarticle IWB-2420; and periodic inspections will be performed in accordance with NUREG-2191, Table XI.M35-1, Category C.

This new aging management program will be implemented no later than six years prior to the second period of extended operation. The one-time inspections are required to be performed within the six years prior to the second period of extended operation, and no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.24 External Surfaces Monitoring of Mechanical Components (3953023-24)

The External Surfaces Monitoring of Mechanical Components aging management program is a new condition monitoring program that will manage loss of material and cracking of metallic components, as well as loss of

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material, cracking, and hardening and loss of strength for elastomeric components, loss of preload for HVAC closure bolting, reduction of heat transfer for heat exchanger external surfaces exposed to air (room cooler air intake screens), and reduced thermal insulation resistance. Periodic visual inspections, not to exceed a refueling outage interval, of metallic components, elastomers, and insulation jacketing (insulation when not jacketed) will be conducted. There are no cementitious components in the scope of this program. This program does not monitor for reduction of heat transfer due to fouling for heat exchanger internal surfaces exposed to air. This aging effect will be managed by the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (R.2.1.25) program. This program does not manage cracking due to stress corrosion cracking (SCC) or loss of material in aluminum and SS components exposed to aqueous solutions and air environments containing halides. As discussed in SLRA Sections 3.1.2.2.16, 3.2.2.2.4, 3.3.2.2.3, 3.4.2.2.2, 3.2.2.2.2, 3.3.2.2.4, 3.4.2.2.3, 3.2.2.2.8, 3.3.2.2.8, 3.4.2.2.7, 3.2.2.2.10, 3.3.2.2.10, and 3.4.2.2.9, these aging effects for these materials and environments are managed by the One-Time Inspection (R.2.1.21) program.

For certain materials, such as flexible polymers and elastomers, physical manipulation to detect hardening or loss of strength will be used to augment the visual examinations conducted under this program. A sample of outdoor component surfaces that are insulated and a sample of indoor insulated components exposed to condensation (due to the in-scope component being operated below the dew point), will be periodically inspected every 10 years during the second period of extended operation. Inspections not conducted in accordance with ASME Code Section XI requirements will be conducted in accordance with plant-specific procedures which include inspection parameters such as lighting, distance, offset, and surface conditions. Acceptance criteria are such that the component will meet its intended function until the next inspection or the end of the second period of extended operation. Qualitative acceptance criteria will be defined to reasonably assure a singular decision is derived based on observed conditions.

This new aging management program will be implemented no later than six months prior to the second period of extended operation.

R.2.1.25 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (3953023-25)

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is a new condition monitoring program that will manage loss of material and cracking of metallic components, as well as loss of material and hardening and loss of strength of elastomeric materials. Reduction of heat transfer and flow blockage will also be managed. This program will consist of visual inspections of all accessible internal surfaces of piping, piping components, ducting, heat exchanger components, and other mechanical components. Applicable environments include condensation, closed cycle cooling water, diesel exhaust, fuel oil, lube oil, raw water, treated water, and waste water. Visual (VT-1) or surface examinations will be performed to detect cracking of stainless steel components exposed to a diesel exhaust

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environment. Visual (VT-1), surface, or volumetric examinations will be performed to detect cracking of titanium components exposed to raw water. Except for hardening and loss of strength of elastomers, aging effects associated with components within the scope of the Open Cycle Cooling Water System (R.2.1.11) program, Closed Treated Water Systems (R.2.1.12) program, and Fire Water System (R.2.1.17) program will not be managed by this program. Loss of material due to recurring internal corrosion on the drain pans of the HPCI, RCIC, Core Spray and RHR pump room unit coolers will be managed by this program. Additionally, in accordance with NUREG-2191, AMP XI.M42, "Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks", loss of coating integrity for certain internally coated tanks in the Radwaste and Reactor Water Cleanup Systems will be performed by this program.

Internal inspections will be performed during the periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. At a minimum, in each 10-year period during the second period of extended operation, a representative sample of 20 percent of the population (defined as components having the same combination of material, environment, and aging effect) or a maximum of 19 components per population per unit, where the sample size is not based on percentage of the population, will be inspected. Sample selection will consider component susceptibility to aging due to factors such as time in service and severity of operating conditions. Opportunistic inspections will continue in each period despite meeting the sampling limit. For certain materials, such as flexible polymers, physical manipulation or pressurization to detect hardening or loss of strength will be used to augment the visual examinations conducted under this program.

Inspections not conducted in accordance with ASME Code Section XI requirements will be conducted in accordance with plant-specific procedures which include inspection parameters such as lighting, distance, offset, and surface conditions. Acceptance criteria will insure that the component will meet its intended function until the next inspection or the end of the second period of extended operation. Qualitative acceptance criteria are clear enough to reasonably assure a singular decision is derived based on observed conditions.

This new aging management program will be implemented no later than six months prior to the second period of extended operation.

R.2.1.26 Lubricating Oil Analysis (3953023-26)

The Lubricating Oil Analysis aging management program is an existing condition monitoring program that provides monitoring of oil in piping, piping components, gear boxes, heat exchangers, and tanks within the scope of license renewal exposed to a lubricating oil environment. The program provides reasonable assurance that the oil environment in the mechanical systems is maintained to the required quality, and the oil systems are maintained free of contaminants (primarily water and particulates), thereby preserving an environment that is not conducive to loss of material or reduction of heat transfer. Program activities include sampling, analysis, and trending of

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lubricating oil for detrimental contaminants. The presence of water or particulates may also indicate in-leakage and corrosion product buildup.

R.2.1.27 Monitoring of Neutron-Absorbing Materials Other Than Boraflex (3953023-27)

The Monitoring of Neutron-Absorbing Materials Other Than Boraflex aging management program is an existing condition monitoring program that includes periodic inspection, testing, monitoring, and analysis of test coupons of the neutron-absorbing material in the spent fuel storage racks to assure that the required five percent sub-criticality margin is maintained. This program consists of inspecting the physical condition of the neutron-absorbing material for visual appearance, dimensional measurements, weight, geometric changes (e.g., bubbling, blistering, corrosion, pitting, cracking, and flaking), and boron areal density as observed from coupons, to monitor for reduction of neutron absorbing capacity, loss of material, and change in dimension. This program is further described in UFSAR Section 10.3.6.2.

R.2.1.28 Buried and Underground Piping and Tanks (3953023-28)

The Buried and Underground Piping and Tanks aging management program is an existing condition monitoring program that manages the aging effects associated with the external surfaces of buried and underground piping and tanks including loss of material and cracking. It addresses piping and tanks composed of any material, including carbon steel and stainless steel.

The program also manages aging through preventive and mitigative actions (i.e., coatings, backfill quality, and cathodic protection). The number of inspections is based on the effectiveness of the preventive and mitigative actions. Annual cathodic protection surveys are conducted. For steel components, where the acceptance criteria for the effectiveness of the cathodic protection is other than -850 mV instant off, loss of material rates are measured.

Inspections are conducted by qualified individuals. Where the coatings, backfill or the condition of exposed piping does not meet acceptance criteria such that the depth or extent of degradation of the base metal could have resulted in a loss of pressure boundary function when the loss of material rate is extrapolated to the end of the second period of extended operation, an increase in the sample size will be conducted. If a reduction in the number of inspections recommended in NUREG-2191, AMP XI.M41, Table XI.M41-2 is claimed based on a lack of soil corrosivity as determined by soil testing, then soil testing is conducted once in each 10-year period starting 10 years prior to the second period of extended operation.

The Buried and Underground Piping and Tanks aging management program will be enhanced to:

1. Manage cracking for buried stainless steel piping, utilizing a method that has been demonstrated to be capable of detecting cracking, whenever coatings are removed exposing the base material.
2. Perform direct visual inspection of buried piping within the scope of license

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renewal in accordance with NUREG-2191, Table XI.M41-2, and sections 4.a and 4.b, during each 10-year period, beginning 10 years prior to the second period of extended operation. The number of inspections of buried piping will be based upon the as-found results of cathodic protection system availability and effectiveness. The length of piping for each inspection will be based on the recommendations in section 4.c.

3. Perform extent of condition inspections as follows: When measured pipe wall thickness, projected to the end of the second period of extended operation, does not meet the minimum pipe wall thickness requirements due to external environments, the number of inspections within the affected piping categories will be doubled or increased by five, whichever is smaller. If adverse indications are found in the expanded sample, an analysis will be conducted to determine the extent of condition and extent of cause. The size of the follow-up inspections will be determined based on the analysis. Timing of any additional inspections will be based on the severity of the identified degradation and the consequences of leakage or loss of function. Any additional inspections will be performed within the same 10-year inspection interval in which the original degradation was identified, or within four years after the end of the 10-year interval if the degradation was identified in the latter half of the 10-year interval. Expansion of sample size may be limited by the extent of piping subject to the observed degradation mechanism or if the piping system or portion of the system is replaced or otherwise mitigated within the same 10-year inspection interval in which the original degradation was identified or within four years after the end of the 10-year interval, if the degradation was identified in the latter half of the 10-year interval.

4. Upgrade existing cathodic protection system no later than 5 years prior to the second period of extended operation in accordance with NACE SP0169-2007 to ensure effective control of external corrosion of underground piping and tanks.

5. Perform examination of buried emergency diesel generator fuel oil tanks from the internal surface of the tank using volumetric techniques during each 10-year period, beginning 10 years prior to the second period of extended operation. A minimum of 25 percent coverage is required.

6. Perform annual system monitoring of the cathodic protection system to ensure effective protection of buried piping.

7. Apply coating to buried portions of the 10-inch diameter stainless steel line from the torus dewatering tank to the condensate transfer pump suction line in accordance with approved station specifications, during the 10-year period prior to the second period of extended operation.

These enhancements will be implemented no later than 10 years prior to the second period of extended operation, unless a more specific schedule is described within the enhancement (i.e., Enhancement 4). Inspections that are required to be performed in the 10-year period prior to the second period of extended operation will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

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R.2.1.29 Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (3953023-29)

The Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks aging management program is a new condition monitoring program that manages degradation of internal coatings/linings exposed to raw water, treated water, waste water, condensation, or lubricating oil that can lead to loss of material of base metals or downstream effects such as reduction in flow, pressure, or heat transfer when coatings/linings become debris. There are no piping or components with internal coatings/linings in the program scope that are exposed to closed-cycle cooling water, treated borated water, or fuel oil. This program is not used to manage the integrity of coatings applied to external surfaces of components.

This program manages these aging effects for internal coatings by conducting periodic visual inspections of all coatings/linings applied to the internal surfaces of in scope components where loss of coating or lining integrity could impact the component's or downstream component's current licensing basis intended function(s). The internal surfaces of the Condensate Storage Tanks and Refueling Water Storage Tank are coated; aging management for these tanks is covered under the Outdoor and Large Atmospheric Metallic Storage Tanks (R.2.1.18) program and includes the applicable requirements for coating inspections from the Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks program. Aging management of galvanized piping in the Plant Equipment and Floor Drain System, and internally coated tanks in the Radwaste and Reactor Water Cleanup Systems will be performed under the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (R.2.1.25) program.

For tanks and heat exchangers, all accessible surfaces are inspected. Piping inspections are sampling-based. Baseline inspections not previously conducted will be performed in the 10-year period prior to the second period of extended operation. For non-cementitious coatings/linings, the training and qualification of individuals involved in coating/lining inspections, and the evaluation of inspection results, are conducted in accordance with ANSI and ASTM International Standards endorsed in RG 1.54, including guidance from the staff associated with a particular standard. For cementitious coatings, training and qualifications are based on an appropriate combination of education and experience related to inspecting concrete surfaces. Peeling and delamination is not acceptable. Blisters are evaluated by a coatings specialist, and should be limited to a few intact small blisters that are completely surrounded by sound material, and with size and frequency not increasing. Minor cracks in cementitious coatings are acceptable provided there is no evidence of debonding. All other degraded conditions are evaluated by a coatings specialist. For coated/lined surfaces determined to not meet the acceptance criteria, physical testing is performed where physically possible (i.e., sufficient room to conduct testing) in conjunction with repair or replacement of the coating/lining.

This new aging management program will be implemented no later than 10 years prior to the second period of extended operation. Baseline inspections

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that may be required in the 10-year period prior to the second period of extended operation will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.30 ASME Section XI, Subsection IWE (3953023-30)

The ASME Section XI, Subsection IWE aging management program is an existing condition monitoring program based on ASME Code and complies with the provisions of 10 CFR 50.55a. The program consists of periodic visual, surface, and volumetric examinations, where applicable, of metallic pressure-retaining components of steel containments for signs of degradation, damage, irregularities, and for coated areas distress of the underlying metal shell, and corrective actions. Acceptability of inaccessible areas of steel containment shell is evaluated when conditions found in accessible areas indicate the presence of, or could result in, flaws or degradation in inaccessible areas.

This program also includes aging management for the potential loss of material due to corrosion in the inaccessible areas of the BWR Mark I steel containment. In addition, the program includes supplemental surface examination to detect cracking for high temperature mechanical penetrations subject to cyclic loading but have no CLB fatigue analysis; and if triggered by plant-specific operating experience, a one-time supplemental volumetric examination by sampling randomly selected as well as focused locations susceptible to loss of thickness due to corrosion of containment shell that is inaccessible from one side. Inspection results are compared with prior recorded results in acceptance of components for continued service.

The ASME Section XI, Subsection IWE aging management program will be enhanced to:

1. Perform surface examinations on accessible portions of high temperature drywell mechanical penetrations, in addition to visual examinations, to detect cracking, once per 10-year interval during the second period of extended operation.
2. Clarify that the recommended guidance for proper selection of bolting material and lubricants, and appropriate installation torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting is a requirement at Peach Bottom in accordance with the guidelines provided in EPRI NP-5067 and TR-104213. Clarify that the recommended requirements for storage, lubricant selection, and bolting and coating material selection include the recommendations in Section 2 of Research Council on Structural Connections (RCSC) publication "Specification for Structural Joints Using High-Strength Bolts," are a requirement at Peach Bottom.
3. Implement a one-time supplemental volumetric examination of the containment metal shell surfaces that are inaccessible from one side, if triggered by plant-specific OE. The trigger for this supplemental examination is plant-specific occurrence or recurrence of measurable metal shell corrosion (base metal material loss exceeding 10 percent of nominal

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plate thickness) initiated on the inaccessible side or areas, identified since the date of issuance of the first renewed license. This supplemental volumetric examination consists of a sample of one-foot square locations that include both randomly-selected and focused areas most likely to experience degradation based on plant-specific OE and/or other relevant considerations such as environment. The sample size, locations, and any needed scope expansion (based on findings) for this one-time set of volumetric examinations should be determined on a plant-specific basis to demonstrate statistically with 95 percent confidence that 95 percent of the accessible portion of the containment liner is not experiencing corrosion degradation with greater than 10 percent loss of nominal thickness.

These enhancements will be implemented no later than six months prior to the second period of extended operation.

R.2.1.31 ASME Section XI, Subsection IWF (3953023-31)

The ASME Section XI, Subsection IWF aging management program is an existing condition monitoring program that consists of periodic visual examinations of ASME Class 1, 2, 3, and MC piping and component supports and high-strength structural bolting for signs of degradation (such as loss of material, loss of mechanical function, cracking, and loss of preload), evaluation, and corrective actions. The program is implemented through corporate and station procedures, in accordance with the requirements of the ASME Code, Section XI, Subsection IWF, as approved in 10 CFR 50.55a. The monitoring methods are effective in detecting the applicable aging effects and the frequency of monitoring is adequate to prevent significant degradation.

This program consists of periodic visual examination of piping and component supports for signs of degradation, evaluation, and corrective actions. The program will be enhanced to implement additional inspections beyond the inspections required by the 10 CFR 50.55a ASME Code Section XI, Subsection IWF program. This consists of a one-time inspection of an additional five percent of the sample size specified in Table IWF-2500-1 for Class 1, 2, and 3 piping supports. This one-time inspection will be conducted within the five years prior to entering the second period of extended operation. For high-strength bolting in sizes greater than 1 inch nominal diameter, volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1 will be performed to detect cracking in addition to the VT-3 examination.

The program will be enhanced to address sampling requirements for elective repairs. If a component support does not exceed the acceptance standards of IWF-3400 but is electively repaired to as-new condition, the sample is increased or modified to include another support that is representative of the remaining population of supports that were not repaired.

The ASME Section XI, Subsection IWF aging management program will be enhanced to:

1. Perform periodic evaluations of the acceptability of inaccessible areas of supports (e.g., portions of supports encased in concrete, buried underground,

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or encapsulated by guard pipe), when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to inaccessible areas of supports. Perform these evaluations once every 10 years during the second period of extended operation.

2. Perform a one-time inspection of an additional five percent of the currently inspected sample size specified in Table IWF-2500-1 for Class 1, 2, and 3 piping supports. Conduct the one-time inspection within the five years prior to entering the second period of extended operation. Select the additional supports from the remaining population of IWF piping supports. Ensure that the sample expansion includes components that are most susceptible to age-related degradation (i.e., based on factors such as time in service, material, and aggressiveness of the environment).

3. Perform VT-3 examinations of all ASTM A-490 bolting materials, used for the reactor vessel support skirts and for the core spray pump supports once per 10-year interval during the second period of extended operation. Perform volumetric examination comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, of 12 ASTM A490 bolts at each of the reactor vessel support skirts, once per 10-year interval during the second period of extended operation. If the volumetric examination of these ASTM A490 bolts reveals conditions that do not meet acceptance criteria, enter the results into the corrective action program and extend the ASTM A490 bolt examination scope to include other ASTM A490 bolts used in similar joint configurations and subject to similar environmental exposure conditions, which is comparable to the methodology used by the ASME Code, section IWF-2430 for IWF component supports.

4. Clarify that the recommended guidance for proper selection of bolting material and lubricants, and appropriate installation torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting is a requirement at Peach Bottom in accordance with the guidelines provided in EPRI NP-5067 and TR-104213. Clarify that the recommended requirements for storage, lubricant selection, and bolting and coating material selection include the recommendations in Section 2 of Research Council on Structural Connections (RCSC) publication "Specification for Structural Joints Using High-Strength Bolts," are a requirement at Peach Bottom.

5. Enhance engineering procedures to require volumetric examination should high-strength bolting (actual measured yield strength greater than or equal to 150 ksi) in sizes greater than 1-inch nominal diameter (including ASTM A490 and equivalent ASTM F2280) be installed. The examination shall be comparable to that of ASME Code Section XI, Table IWB-2500-1, Examination Category B-G-1, at least once per 10-year interval, to detect cracking, in addition to the VT-3 examination.

6. Provide guidance, regarding the selection of supports to be inspected on subsequent inspections, when a support that does not meet the threshold of "unacceptable for continued service" as defined in IWF-3400, is restored in accordance with the Corrective Action Program. The enhanced guidance will ensure that the sample is increased or modified to include another support that

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is representative of the remaining population of supports that were not repaired.

These enhancements will be implemented in accordance with the schedule described within the enhancements. Inspections that are required to be performed in the five-year period prior to the second period of extended operation will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.32 10 CFR Part 50, Appendix J (3953023-32)

The 10 CFR 50 Appendix J aging management program is an existing condition monitoring program that consists of monitoring leakage rates through the containment system, its shell, associated welds, penetrations, isolation valves, fittings, and other access openings to detect degradation of the containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. Consistent with the current licensing basis, the containment leak rate tests are performed in accordance with the regulations and guidance provided in 10 CFR Part 50 Appendix J, Option B, NEI 94-01, Revision 2-A and Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J", and subject to the requirements of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants".

R.2.1.33 Masonry Walls (3953023-33)

The Masonry Walls aging management program is an existing condition monitoring program that consists of visual inspections, based on IE Bulletin 80-11 and plant-specific monitoring proposed by IN 87-67, for managing shrinkage, separation, gaps, loss of material, and cracking of masonry walls such that the evaluation basis is not invalidated and intended functions are maintained.

The Masonry Walls aging management program will be enhanced to:

1. Expand the program to include masonry walls in the Administration Building and Dewatering Building.

This enhancement will be implemented no later than six months prior to the second period of extended operation.

R.2.1.34 Structures Monitoring (3953023-34)

The Structures Monitoring aging management program is an existing condition monitoring program that consists of periodic visual inspection and monitoring of the condition of concrete and steel structures, structural components, component supports, and structural commodities to ensure that aging degradation (such as those described in ACI 349.3R, ACI 201.1R, SEI/ASCE 11, and other documents) will be detected, the extent of degradation determined and evaluated, and corrective actions taken prior to loss of intended functions. Structures are monitored on an interval not

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to exceed 5 years. Inspections also include seismic joint fillers, elastomeric materials; and steel edge supports and steel bracings associated with masonry walls, and periodic evaluation of groundwater chemistry and opportunistic inspections for the condition of below grade concrete. Quantitative results (measurements) and qualitative information from periodic inspections are trended with sufficient detail, such as photographs and surveys for the type, severity, extent, and progression of degradation, to ensure that corrective actions can be taken prior to a loss of intended function. The acceptance criteria are derived from applicable consensus codes and standards. For concrete structures, the program includes personnel qualifications and quantitative evaluation criteria of ACI 349.3R.

The Structures Monitoring aging management program will be enhanced to:

1. Explicitly include the following components and commodities within the scope of the program:
 - a. Bearing pads for supports
 - b. Electrical duct banks
 - c. Electrical raceway such as cable tray, conduit, and wireway gutter
 - d. Hatches and plugs
 - e. Manholes and handholes
 - f. Miscellaneous components such as louvers
 - g. Panels, racks, frames, cabinets, and other enclosures
 - h. Permanent shielding blankets
2. Add the following structures to the scope of the program:
 - a. Administration Building
 - b. Boiler House
 - c. Dewatering Building
3. Perform inspections under the enhanced program in order to establish quantitative baseline inspection data prior to the second period of extended operation.
4. Provide evaluation criteria for structural concrete using quantitative second tier criteria of Chapter 5 in ACI 349.3R.
5. Monitor for reduction in concrete anchor capacity if local concrete degradation such as cracking and loss of material is identified.
6. Develop a new implementing procedure or revise an existing implementing procedure to address aging management of inaccessible areas exposed to potentially aggressive groundwater/soil environment that will include the following:
 - a. Monitor raw water and ground water chemistry, for pH, chlorides, and sulfates, on a frequency not to exceed five years that accounts for

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seasonal variations (e.g., quarterly monitoring every fifth year), from locations that are representative of the groundwater in contact with structures within the scope of second license renewal.

- b. Enter adverse results, which exceed water chemistry criteria, into the corrective action program. As part of the corrective actions, if aggressive groundwater is identified that might affect structures in scope for license renewal, perform additional water testing at additional locations and perform soil testing in order to confirm the extent, severity, and potential aging mechanisms resulting from the aggressive groundwater/soil.
 - c. Develop engineering evaluations to evaluate the water chemistry results to assess the impact, if any, on below-grade concrete, including the potential for further degradation due to the aggressive groundwater, as well as consideration of current conditions. As part of the engineering evaluations, determine if additional actions are warranted, which might include enhanced inspection techniques and/or increased frequency, destructive testing, and focused inspections of representative accessible (leading indicator) or below grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil.
 - d. Develop the initial engineering evaluations prior to the second period of extended operation. Develop follow-up engineering evaluations on an interval not to exceed five years.
 - e. If aggressive groundwater and soil is identified, at a minimum, perform focused inspections of representative, accessible (leading indicator) structural elements, or if accessible areas will not be leading indicators for the potential aging mechanisms, excavate and inspect buried concrete elements exposed to aggressive groundwater/soil.
 - f. If degraded concrete is identified, as part of the focused inspections of leading indicators (representative, accessible or exposed inaccessible concrete), enter adverse results that exceed ACI 349.3R tier 2 criteria into the corrective action program, and expose inaccessible concrete so that the extent of the condition can be determined, baseline conditions documented, and additional actions identified such as repairs, new preventative actions, additional evaluations, and future inspections.
7. Monitor and trend through-wall groundwater leakage, infiltration volumes, and leakage water chemistry for signs of concrete or steel reinforcement degradation. Develop additional engineering evaluations, which consider more frequent inspections, as well as destructive testing of affected concrete to validate existing concrete properties, and leakage water chemistry results. If leakage volumes allow, consider water chemistry analysis of the leakage pH, along with mineral, chloride, sulfate and iron content in the water.
8. Expand the program to monitor accessible sliding surfaces for indications of significant loss of material due to wear or corrosion, and for accumulation of debris or dirt. Establish acceptance criteria for sliding surfaces as no significant loss of material due to wear or corrosion, and no debris or dirt that

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could restrict or prevent sliding of the surfaces, as required by design.

9. Evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas

10. Expand the program to monitor elastomeric vibration isolators and bearing pads for cracking, loss of material, and hardening. Supplement visual inspection of elastomeric elements with tactile inspection to detect hardening, if the intended function is suspect. Establish acceptance criteria for elastomeric pads and vibration isolation elements as no loss of material, cracking, or hardening that can lead to loss of isolation or support function.

11. Clarify that loose bolts and nuts and cracked bolts are not acceptable unless accepted by engineering evaluations.

12. Expand the program to inspect the fiberglass outer covering of permanent shielding blankets for signs of tears. If a tear is found, enter the condition into the corrective action program for evaluation. Repair or replace the permanent shielding, unless an evaluation determines that the condition is acceptable.

13. Clarify that the recommended guidance for proper selection of bolting material and lubricants, and appropriate installation torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting is a requirement at Peach Bottom in accordance with the guidelines provided in EPRI NP-5067 and TR-104213. Clarify that the recommended requirements for storage, lubricant selection, and bolting and coating material selection include the recommendations in Section 2 of Research Council on Structural Connections (RCSC) publication "Specification for Structural Joints Using High-Strength Bolts," are a requirement at Peach Bottom.

These enhancements will be implemented no later than six months prior to the second period of extended operation. Baseline inspections will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.35 Inspection of Water-Control Structures Associated with Nuclear Power Plants (3953023-35)

The Inspection of Water-Control Structures Associated with Nuclear Power Plants aging management program is an existing condition monitoring program that consists of inspection and surveillance of the raw water control structures associated with emergency cooling systems, which are the Circulating Water Pump Structure and the Emergency Cooling Tower and Reservoir. The PBAPS design does not include dams, slopes, canals, and other raw water-control structures associated with emergency cooling water systems or flood protection in the scope of this program. The program includes reinforced concrete, structural steel, and structural bolting associated with the water-control structures. The program also includes PVC drift eliminators, ceramic tile fill, and cast iron fill supports at the Emergency Cooling Tower and Reservoir. In general, parameters monitored are in accordance with Section

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C.2 of RG 1.127 and quantitative measurements are recorded for findings that exceed the acceptance criteria for applicable parameters monitored or inspected. Inspections are performed at least once every five years for structural components that are not submerged. Submerged components in the Circulating Water Pump Structure are inspected at least once every six years. Submerged components in the Emergency Cooling Tower and Reservoir are inspected at least once every 10 years. Structures exposed to aggressive water require additional plant-specific investigation.

Not included in this program is the offsite Conowingo Hydroelectric Plant (Dam), which is operated under a separate license and is subject to the FERC five-year inspection program. Aging management for the dam includes activities such as visual inspections by a qualified independent consultant approved by FERC, and submittal of inspection reports with corrective actions that are approved by FERC. Aging management for the dam is performed in accordance with FERC requirements, 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 inspections performed under the FERC five-year inspection program, called the FERC Inspections of the Conowingo Hydroelectric Plant (Dam) (Appendix R.5, Commitment 50), have been accepted by FERC, and are the current licensing basis for PBAPS regarding aging management of the dam. PBAPS will continue to comply with these FERC requirements during the second period of extended operation.

The Inspection of Water-Control Structures Associated with Nuclear Power Plants aging management program will be enhanced to:

1. Explicitly include the sluice gates at the Circulating Water Pump Structure within the scope of the program.
2. Clarify parameters to be monitored and inspected at the Emergency Cooling Tower and Reservoir to include visual inspection for loss of material and reduction of heat transfer due to fouling for the cooling tower fill, and visual inspection of the drift eliminators.
3. Monitor for reduction in concrete anchor capacity if local concrete degradation such as cracking and loss of material is identified.
4. Expand the program to monitor accessible sliding surfaces for indications of significant loss of material due to wear or corrosion, and for accumulation of debris or dirt.
5. Include provisions for special inspections following significant natural phenomena, such as large floods, hurricanes, tornadoes, or intense local rainfall as part of the guidelines for severe weather and natural disasters.
6. Develop a new implementing procedure or revise an existing implementing procedure to address aging management of inaccessible areas exposed to potentially aggressive groundwater/soil environment that will include the following:

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- a. Monitor raw water and ground water chemistry, for pH, chlorides, and sulfates, on a frequency not to exceed five years that accounts for seasonal variations (e.g., quarterly monitoring every fifth year), from locations that are representative of the groundwater in contact with structures within the scope of second license renewal.
 - b. Enter adverse results, which exceed water chemistry criteria, into the corrective action program. As part of the corrective actions, if aggressive groundwater is identified that might affect structures in scope for license renewal, perform additional water testing at additional locations and perform soil testing in order to confirm the extent, severity, and potential aging mechanisms resulting from the aggressive groundwater/soil.
 - c. Develop engineering evaluations to evaluate the water chemistry results to assess the impact, if any, on below-grade concrete, including the potential for further degradation due to the aggressive groundwater, as well as consideration of current conditions. As part of the engineering evaluations, determine if additional actions are warranted, which might include enhanced inspection techniques and/or increased frequency, destructive testing, and focused inspections of representative accessible (leading indicator) or below grade, inaccessible concrete structural elements exposed to aggressive groundwater/soil.
 - d. Develop the initial engineering evaluations prior to the second period of extended operation. Develop follow-up engineering evaluations on an interval not to exceed five years.
 - e. If aggressive groundwater and soil is identified, at a minimum, perform focused inspections of representative, accessible (leading indicator) structural elements, or if accessible areas will not be leading indicators for the potential aging mechanisms, excavate and inspect buried concrete elements exposed to aggressive groundwater/soil.
 - f. If degraded concrete is identified, as part of the focused inspections of leading indicators (representative, accessible or exposed inaccessible concrete), enter adverse results that exceed ACI 349.3R tier 2 criteria into the corrective action program, and expose inaccessible concrete so that the extent of the condition can be determined, baseline conditions documented, and additional actions identified such as repairs, new preventative actions, additional evaluations, and future inspections.
7. Monitor and trend through-wall groundwater leakage, infiltration volumes, and leakage water chemistry for signs of concrete or steel reinforcement degradation. Develop additional engineering evaluations, which consider more frequent inspections, as well as destructive testing of affected concrete to validate existing concrete properties, and leakage water chemistry results. If leakage volumes allow, consider water chemistry analysis of the leakage pH, along with mineral, chloride, sulfate and iron content in the water.
8. Evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to

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such inaccessible areas.

9. Document the concrete conditions of submerged concrete structures.
10. Specify a six-year frequency for the inspection of the submerged portions of the traveling screen bays to match the inspection frequency of the submerged portions of the Circulating Water Pump Structure bays.
11. Perform inspections under the enhanced program in order to establish quantitative baseline inspection data prior to the second period of extended operation.
12. Provide evaluation criteria for structural concrete using quantitative second tier criteria of Chapter 5 in ACI 349.3R.
13. Clarify that loose bolts and nuts and cracked bolts are not acceptable unless accepted by engineering evaluations.
14. Clarify that the recommended guidance for proper selection of bolting material and lubricants, and appropriate installation torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting is a requirement at Peach Bottom in accordance with the guidelines provided in EPRI NP-5067 and TR-104213. Clarify that the recommended requirements for storage, lubricant selection, and bolting and coating material selection include the recommendations in Section 2 of Research Council on Structural Connections (RCSC) publication "Specification for Structural Joints Using High-Strength Bolts," are a requirement at Peach Bottom.

These enhancements will be implemented no later than six months prior to the second period of extended operation. Baseline inspections will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.36 Protective Coating Monitoring and Maintenance (3953023-36)

The Protective Coating Monitoring and Maintenance aging management program is an existing mitigative and condition monitoring program that manages the effects of loss of coating integrity of Service Level I coatings, as defined in RG 1.54, Revision 1 or latest revision, inside primary containment. The program manages coating system selection, application, visual inspections, assessments, repairs, and maintenance of Service Level I protective coatings. The program is comparable to a monitoring and maintenance program for Service Level I protective coatings as described in RG 1.54, Revision 2.

Maintenance of Service Level I coatings applied to carbon steel and concrete surfaces inside containment (e.g., steel containment and torus shell, structural steel, supports, penetrations, and concrete walls and floors) will serve to prevent or minimize the loss of material of carbon steel components due to corrosion and aids in decontamination, but these coatings are not credited for managing the effects of corrosion for the carbon steel containment shells and

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components. This program ensures that the Service Level I coatings maintain adhesion so as to not affect the intended function of the emergency core cooling systems (ECCS) suction strainers.

The program also provides controls over the amount of unqualified coatings. Unqualified coating may fail in a way to affect the intended function of the ECCS suction strainers. Therefore, the quantity of degraded and unqualified coating is controlled and assessed periodically to ensure that the amount of unqualified coating in the primary containment is kept within acceptable design limits to support the post-accident operability of the ECCS.

The Protective Coating Monitoring and Maintenance aging management program will be enhanced to:

1. Use Level II or Level III coating inspectors, certified to ANSI N45.2.6, for inspection of Service Level I coatings.

This enhancement will be implemented no later than six months prior to the second period of extended operation.

R.2.1.37 Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (3953023-37)

The Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is an existing condition monitoring program that manages the effects of reduced insulation resistance of electrical insulation for license renewal in scope, non-EQ, electrical cables and connections during the second period of extended operation.

In most areas of PBAPS, the actual ambient environments (e.g., temperature, radiation, or moisture) are less severe than the plant design environment. An adverse localized environment (ALE) is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable or connection. Electrical insulation used in electrical cables and connections may degrade more rapidly than expected in these adverse localized environments.

Accessible cables and connections located in adverse localized environments are managed by visual inspection. These cables and connections are visually inspected at least once every 10 years for cable jacket and connection insulation surface anomalies, such as embrittlement, discoloration, cracking, melting, swelling, or surface contamination that could indicate incipient conductor insulation aging degradation from temperature, radiation, or moisture. This is an adequate inspection frequency to preclude failures of the cable and connection insulation since experience shows that aging degradation is a slow process.

Additional inspections, repairs, or replacements are initiated as appropriate under the corrective action program. If visual inspections identify degraded or damaged conditions that may impact the cable system's ability to perform its intended functions, then testing may be performed for evaluation. Testing may

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include thermography and one or more proven condition monitoring test methods applicable to the cable and connection insulation material. Testing as part of an existing maintenance, calibration or surveillance program may be credited. Electrical cable and connection insulation material test results are to be within the acceptance criteria, as identified in the station's procedures.

The Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will be enhanced to:

1. Include potential follow-up actions when visual inspections identify degraded or damaged conditions that may impact the performance of intended functions:
 - a. Perform tests, for condition monitoring when visual inspections identify damaged or degraded insulation of in scope cables and connections. When a large number of cables are identified as damaged or degraded, a sample population will be tested. The sample size will be 20 percent of each affected cable and connection type with a maximum sample size of 25.
 - b. Document the basis for the sample selected for testing when visual inspections identify damaged or degraded insulation conditions for in scope cables and connections.
2. Visually inspect and evaluate cables and connections that were exposed to adverse localized environments (ALEs), which have since been mitigated, on an at least once every 10-year frequency, to assure the cumulative aging effects for electrical insulation, in remedied ALEs are not impacting the ongoing ability of the cables and connections to perform their intended function during the second period of extended operation.

These enhancements will be implemented no later than six months prior to the second period of extended operation. In addition, the first inspections incorporating enhancements will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.38 Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (3953023-38)

The Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits aging management program is an existing performance monitoring program that will manage the effects of reduced insulation resistance of non-EQ cable and connection electrical insulation in instrumentation circuits with sensitive, high voltage, low-level current signals. The program applies to the in scope portions of the Neutron Monitoring System and the Radiation Monitoring System, that are located in areas with potential adverse environments and are not managed by the Environmental Qualification of Electric Equipment (R.3.1.3) program. The adverse localized environments are caused by temperature, radiation, or moisture. These adverse localized

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environments can result in reduced insulation resistance causing increases in leakage currents. Other instrument circuits in the Neutron Monitoring System and Radiation Monitoring System are not in scope of this aging management program either because they do not perform a license renewal intended function; they are not sensitive high voltage, low-level signal circuits; or they are managed by the Environmental Qualification of Electric Equipment (R.3.1.3) program.

Calibration or surveillance testing will be performed for the in scope circuits when the cables are included as part of the calibration or surveillance circuit. The calibration and surveillance results will be periodically reviewed to provide an indication of the existence of aging effects based on acceptance criteria for instrumentation circuit performance. Review of results obtained during normal calibration and surveillance may detect severe aging degradation prior to the loss of the cable and connection intended function. A proven cable test (such as insulation resistance tests, time domain reflectometry tests, or other testing judged to be effective in determining cable system insulation condition) will be performed for the in scope circuits when the cables are not included as part of the calibration or surveillance or as an alternative to the review of calibration or surveillance results.

Periodic review of calibration or surveillance results will first be performed prior to the second period of extended operation and at least once every 10 years during the second period of extended operation. If cable tests are credited for program implementation, cable test frequency will be based on engineering evaluation and will be performed at least once every 10 years.

The Electrical Insulation for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits aging management program will be enhanced to:

1. Add the following radiation monitors to the scope of this program
 - a. Main steam line radiation monitors
 - b. Reactor building ventilation exhaust radiation monitors
 - c. Control room fresh air supply radiation monitors
 - d. Control room emergency ventilation supply radiation monitors
 - e. Main stack radiation monitors.
2. Revise the implementing procedures to include documented periodic review of calibration test results for neutron monitors and radiation monitors within the scope of this program. Perform the first periodic review for second license renewal prior to the second period of extended operation and at least every 10 years thereafter.

These enhancements will be implemented no later than six months prior to the second period of extended operation. The first documented periodic review will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

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R.2.1.39 Electrical Insulation for Inaccessible Medium-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (3953023-39)

The Electrical Insulation for Inaccessible Medium-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is an existing condition monitoring program that will manage the effects of reduced insulation resistance of non-EQ, in scope, inaccessible (e.g., installed in buried conduits, cable trenches, cable troughs, duct banks, underground vaults, or direct buried installations), medium voltage power cables (operating voltage; 2 kV to 35 kV), exposed to significant moisture. For this program, significant moisture is defined as exposure to moisture that lasts more than three days (i.e., long term wetting or submergence over a continuous period) that if left unmanaged, could potentially lead to a loss of intended function.

The cables within the scope of this program will be tested using one or more proven tests for detecting reduced insulation resistance of the cable's insulation system due to wetting or submergence, such as dielectric loss (dissipation factor or power factor), AC voltage withstand, partial discharge, step voltage, time domain reflectometry, insulation resistance and polarization index, line resonance analysis, or other testing that is state of the art at the time the test is performed. The first tests will be completed prior to the second period of extended operation. The cables will be tested at least once every six years thereafter. More frequent testing may occur based on test results and operating experience.

Submarine or other cables designed for continuous wetting or submergence are also included in this program as a one-time test with additional periodic tests and inspections determined by one-time inspection results and industry and plant-specific operating experience.

Periodic inspections are performed to prevent inaccessible cable from being exposed to significant moisture such as identifying and inspecting in scope accessible cable conduit ends and cable manholes/vaults for water accumulation, and subsequent draining of accumulated water or other corrective actions, as needed. Prior to the second period of extended operation, the frequency of inspections for accumulated water will be established and adjusted based on plant-specific operating experience with cable wetting or submergence, including water accumulation over time and event driven occurrences such as heavy rain or flooding. The first inspections will be completed prior to the second period of extended operation. During the second period of extended operation, the inspections will occur either at least once per year or at least once every five years for manholes with level monitoring, alarms, and subsequent pump-out prior to wetting or submergence of cables.

The Electrical Insulation for Inaccessible Medium-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will be enhanced to:

1. Add periodic cable testing for additional circuits.

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2. Perform cable testing of the circuits in the scope of this program at a frequency of at least once every six years.
3. Add periodic condition monitoring, as a preventive action, for manholes.

These enhancements will be implemented no later than six months prior to the second period of extended operation. Tests and inspections that are required to be performed prior to the second period of extended operation will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.40 Electrical Insulation for Inaccessible Instrument and Control Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (3953023-40)

The Electrical Insulation for Inaccessible Instrument and Control Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new condition monitoring program that will manage the effects of reduced insulation resistance of non-EQ, in scope, inaccessible (e.g., installed in buried conduits, cable trenches, cable troughs, duct banks, underground vaults, or direct buried installations), instrument and control cables, exposed to significant moisture. For this program, significant moisture is defined as exposure to moisture that lasts more than three days (i.e., long term wetting or submergence over a continuous period) that if left unmanaged, could potentially lead to a loss of intended function.

A sample of in scope, inaccessible instrument and control cables, potentially exposed to significant moisture, will be tested in a one-time confirmatory effort. The cables within the scope of this program will be tested using one or more proven tests for detecting reduced insulation resistance. The tests will be performed prior to the second period of extended operation. The in scope inaccessible instrument and control cables, potentially exposed to significant moisture, will also be visually inspected to identify if observable age degradation of the electrical insulation is occurring. The visual inspection will be performed for cables and connections that are accessible during manhole inspections. The visual inspections will occur at least once every six years coincident with manhole inspections that are being performed at least once every year or at least once every five years. The first visual inspection of inaccessible instrument and control cables will be completed prior to the second period of extended operation.

There are no current submarine or other cables designed for continuous wetting or submergence in the scope of this program. Future cables of this design would be considered for inclusion in this program.

Periodic inspections are performed to prevent inaccessible cable from being exposed to significant moisture such as identifying and inspecting in scope accessible cable conduit ends and cable manholes/vaults for water accumulation, and subsequent draining of accumulated water or other corrective actions, as needed. Prior to the second period of extended operation, the frequency of inspections for accumulated water will be established and adjusted based on plant-specific operating experience with

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cable wetting or submergence, including water accumulation over time and event driven occurrences such as heavy rain or flooding. The first inspections will be completed prior to the second period of extended operation. During the second period of extended operation, the manhole inspections will be performed either at least once every year or at least once every five years based on operating experience of individual manhole level monitoring and pump out activities.

This new aging management program will be implemented no later than six months prior to the second period of extended operation. One-time cable testing, initial manhole inspections, and initial visual cable inspections will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.41 Electrical Insulation for Inaccessible Low-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (3953023-41)

The Electrical Insulation for Inaccessible Low-Voltage Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new condition monitoring program that will manage the effects of reduced insulation resistance of non-EQ, in scope, inaccessible (e.g., installed in buried conduits, cable trenches, cable troughs, duct banks, underground vaults, or direct buried installations), low voltage power cables (operating voltage less than 2 kV), exposed to significant moisture. For this program, significant moisture is defined as exposure to moisture that lasts more than three days (i.e., long term wetting or submergence over a continuous period) that if left unmanaged, could potentially lead to a loss of intended function.

In scope, inaccessible low voltage power cables, potentially exposed to significant moisture, will be tested in a one-time confirmatory effort. The cables within the scope of this program will be tested using one or more proven tests for detecting reduced insulation resistance. The tests will be performed prior to the second period of extended operation. The in scope inaccessible low voltage power cables, potentially exposed to significant moisture, will also be visually inspected to identify if observable age degradation of the electrical insulation is occurring. The visual inspection will be performed for cables and connections that are accessible during manhole inspections. The visual inspections will occur at least once every six years co-incident with manhole inspections that are being performed at least once every year or at least once every five years. The first visual inspection of inaccessible low voltage power cables will be completed prior to the second period of extended operation.

There are no current submarine or other cables designed for continuous wetting or submergence in the scope of this program. Future cables of this design would be considered for inclusion in this program.

Periodic actions are performed to prevent inaccessible cable from being exposed to significant moisture such as identifying and inspecting in scope accessible cable conduit ends and cable manholes/vaults for water

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accumulation, and subsequent draining of accumulated water or other corrective actions, as needed. Prior to the second period of extended operation, the frequency of inspections for accumulated water will be established and adjusted based on plant-specific operating experience with cable wetting or submergence, including water accumulation over time and event driven occurrences such as heavy rain or flooding. The first inspections will be completed prior to the second period of extended operation. During the second period of extended operation, the manhole inspections will be performed either at least once every year or at least once every five years based on operating experience of individual manhole level monitoring and pump out activities.

This new aging management program will be implemented no later than six months prior to the period of extended operation. One-time cable testing, initial manhole inspections, and initial visual cable inspections will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.42 Metal Enclosed Bus (3953023-42)

The Metal Enclosed Bus aging management program is a new condition monitoring program that uses sampling and will manage the identified aging effects of in scope metal enclosed bus. The internal portions of the accessible bus enclosure assemblies will be visually inspected for age-related degradation, including cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. The bus insulation will be visually inspected for signs of reduced insulation resistance, such as embrittlement, cracking, chipping, melting, discoloration, swelling, or surface contamination which may indicate overheating or aging degradation. The internal bus insulating supports will be visually inspected for structural integrity and signs of cracks. External surfaces in an air-outdoor environment will be visually inspected for loss of material due to general, pitting, and crevice corrosion. Accessible enclosure assembly elastomers will be visually inspected for age-related degradation, including surface cracking, crazing, scuffing, dimensional change (e.g. “ballooning” and “necking”), shrinkage, discoloration, hardening, and loss of strength.

A sample of accessible bolted connections will be inspected for increased resistance of connection by either performing thermography or measuring the connection resistance using a micro-ohmmeter. In addition to thermography or resistance measurement, bolted connections not covered with heat shrink tape or boots are visually inspected for increased resistance of connection (e.g., loose or corroded bolted connection and hardware including cracked or split washers).

The inspections and resistance measurements will be performed prior to second period of extended operation and at least once every 10 years for indications of aging degradation.

This new aging management program will be implemented no later than six

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months prior to the second period of extended operation. Initial inspections and resistance measurements will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.1.43 Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (3953023-43)

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new condition monitoring program that consists of a representative sample of electrical connections tested prior to the second period of extended operation. The results will be evaluated to determine if there is a need for subsequent periodic testing on a 10-year frequency. The program applies to electrical connections within the scope of second license renewal that are not subject to EQ requirements. The following factors are considered for sampling: voltage level (medium and low-voltage), circuit loading (high loading), connection type, and location (high temperature, high humidity, vibration, etc.). Twenty percent of each connector type population with a maximum sample of 25 constitutes a representative connector sample size. The specific type of test to be performed shall be a proven method for detecting increased resistance of the connection.

The program does not implement visual inspections of cable connection insulation materials as an alternative to thermography.

This new aging management program will be implemented no later than six months prior to the second period of extended operation. Testing and evaluation of results will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

R.2.2 Plant-Specific Aging Management Programs

This section provides summaries of the plant-specific programs credited for managing the effects of aging.

R.2.2.1 Wooden Pole (3953023-44)

The Wooden Pole aging management program is an existing condition monitoring program that manages aging effects associated with the wooden pole adjacent to the Susquehanna Substation. The wooden pole provides the structural support for the conductors connecting the substation to the submarine cable for the alternate AC power for PBAPS's station blackout coping period. There are no preventive or mitigative actions associated with this program. The program manages loss of material and change in material properties by conducting periodic inspections in accordance with corporate specifications of the wooden pole within the scope of the program. Periodic inspections are conducted on a 10-year frequency. Inspection activities consist of visual inspections, sounding, boring, and excavation, as required, to ensure an adequate examination. Acceptance criteria outlined in the corporate specifications ensure appropriate corrective actions are taken based on

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observed conditions. If an inspection identifies a degraded condition associated with the wooden pole, the corrective action program is utilized to facilitate repair or replacement activities.

The Wooden Pole aging management program will be enhanced to:

1. Document results that do not meet the acceptance criteria in the corrective action program.

This enhancement will be implemented no later than six months prior to the second period of extended operation.

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R.3.0 NUREG-2191 CHAPTER X AGING MANAGEMENT PROGRAMS

R.3.1 Evaluation of Chapter X Aging Management Programs

Aging Management Programs evaluated in Chapter X of NUREG-2191 are associated with Time-Limited Aging Analysis for metal fatigue of the reactor coolant pressure boundary and environmental qualification (EQ) of electric components. These programs are evaluated in this section.

R.3.1.1 Fatigue Monitoring (3953023-45)

The Fatigue Monitoring aging management program is an existing preventive program that manages fatigue damage of the reactor pressure vessel components, reactor coolant pressure boundary piping components, and other components per 10 CFR 54.21(c)(1)(iii). The program monitors and tracks the number of occurrences and severity of design basis transients assessed in the applicable fatigue or cyclical loading analyses, including those in applicable cumulative usage fatigue (CUF) analyses and environmental-assisted cumulative usage fatigue (CUF_{en}) analyses. No PBAPS ANSI B31.1 and ASME Code Class 2 and 3 maximum allowable stress range reduction/expansion stress analyses have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), therefore this program does not apply to these implicit analyses. No ASME Section III fatigue waiver analyses have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), therefore this program does not apply to fatigue waiver analyses. No cycle-based flaw growth, flaw tolerance, or fracture mechanics analyses that are based on cycle-based loading assumptions have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), therefore this program does not apply to flaw growth, flaw tolerance, or fracture mechanics analyses. The program also monitors applicable design transient parameters (e.g., temperatures, pressures, displacements, strains, flow rates, etc.) for components with stress-based fatigue calculations.

The program manages cumulative fatigue damage or cracking induced by fatigue or cyclic loading in the applicable structures and components by monitoring for design transient cycles and the calculating CUF values and CUF_{en} values. The program also monitors applicable plant-specific parameters (e.g., temperatures, pressures, flow rates, etc.) used in stress-based fatigue analysis methodologies. No cycle-based flaw growth, flaw tolerance, or fracture mechanics analyses that are based on cycle-based loading assumptions have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), therefore this program does not apply to flaw growth, flaw tolerance, or fracture mechanics analyses. The program utilizes applicable acceptance criteria (limits) to verify the continued acceptability of existing analyses through transient cycle counting and the calculation, trending, and projection of CUF and CUF_{en} values. The program verifies the continued acceptability of existing analyses with periodically updated evaluations of the analyses to demonstrate that they continue to meet the appropriate limits.

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When a program acceptance criterion is exceeded or the severity an actual transient exceeds the design transient definition the condition is entered into the corrective action program and appropriate corrective actions, such as reanalysis, component or structure inspections, or component or structure repair or replacement activities are implemented to ensure that design limits are not exceeded. The program manages cumulative fatigue damage in accordance UFSAR Section 4.2.5 and Tech Spec Section 5.5.5.

The Fatigue Monitoring aging management program will be enhanced to:

1. Update the SI:FatiguePro™ software to include the calculation and tracking of Environmentally Assisted Fatigue (EAF) in accordance NUREG/CR-6909, Revision 1.
2. Update applicable fatigue analyses and monitored component locations based on operating experience, plant modifications, inspection findings, changes to transient definitions, and unanticipated newly discovered fatigue loading events.
3. Provide procedural direction to require periodic validation of chemistry parameters used to determine Fen factors used in SI:FatiguePro™.
4. Provide procedural direction to add an additional acceptance criterion associated with HELB exclusion criteria.

These enhancements will be implemented no later than six months prior to the second period of extended operation.

R.3.1.2 Neutron Fluence Monitoring (3953023-46)

The Neutron Fluence Monitoring aging management program is an existing condition monitoring program that monitors and tracks increasing neutron fluence (integrated, time-dependent neutron flux exposures) to reactor pressure vessel and reactor internal components to ensure that applicable reactor pressure vessel neutron irradiation embrittlement analyses (i.e., TLAAs) and radiation-induced aging effect assessment for reactor internal components will remain within their applicable limits. The components evaluated by these analyses are the reactor pressure vessel shell and welds and reactor vessel internal components subject to reactor coolant and neutron flux environment which are fabricated from carbon or low alloy steel with stainless steel cladding, stainless steel, and nickel alloy materials.

The program has two aspects, one to verify the continued acceptability of existing analyses through neutron fluence monitoring and the other to provide periodically updated evaluations of the analyses involving neutron fluence inputs to demonstrate that they continue to meet the appropriate limits defined in the current licensing basis (CLB).

Monitoring is performed to verify the adequacy of neutron fluence projections, which are defined for the CLB in NRC approved reports. The methods and assumptions for projecting RPV neutron fluence for the beltline region are consistent with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide

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(RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The methods and assumptions used for the original beltline region are considered appropriate for the beltline region that has been extended to encompass materials projected to experience fluence in excess of 1×10^{17} n/cm² (E > 1 MeV) at 70 EFPY, since the extended region does not extend significantly above or below the active fuel region and no additional reactor vessel plate materials (heat numbers) or welds are projected to experience fluence in excess of 1×10^{17} n/cm² (E > 1 MeV). The methods for projecting reactor vessel internal component fast neutron fluence values are not governed by regulatory guidance or requirements.

The program results are compared to the neutron fluence parameter inputs used in the neutron embrittlement analyses for reactor pressure vessel components. This includes but is not limited to the neutron fluence inputs for the reactor pressure vessel upper-shelf energy analyses (or equivalent margin analyses, as applicable to the CLB) and pressure-temperature limits analyses that are required to be performed in accordance in 10 CFR Part 50, Appendix G requirements. Comparisons to the neutron fluence inputs for other analyses (as applicable to the CLB) include those for mean RT_{NDT} and probability of failure analyses for BWR reactor pressure vessel circumferential and axial shell welds, BWR core reflood design analyses, and aging effect assessments for BWR reactor internals that are induced by neutron irradiation exposure mechanisms.

Reactor vessel surveillance capsule dosimetry data obtained in accordance with 10 CFR Part 50, Appendix H requirements and through implementation of the Reactor Vessel Material Surveillance (R.2.1.20) program provide inputs to and have impacts on the neutron fluence monitoring results that are tracked by this program. In addition, regulatory requirements in the plant technical specifications or in specific regulations of 10 CFR Part 50 may apply, including those in 10 CFR Part 50, Appendix G and 10 CFR 50.55a.

The Neutron Fluence Monitoring aging management program will be enhanced to:

1. Perform periodic monitoring of reactor pressure vessel and reactor vessel internals accumulated neutron fluence, every refueling cycle, to ensure that neutron fluence projections used to support reactor pressure vessel neutron irradiation embrittlement analyses (i.e., TLAAs, pressure-temperature limits) and reactor vessel internals aging effect assessments remain bounding with respect to actual plant operating conditions.

This enhancement will be implemented no later than six months prior to the second period of extended operation.

R.3.1.3 Environmental Qualification of Electric Equipment (3953023-47)

The Environmental Qualification of Electric Equipment aging management preventive program is an existing program that ensures maintenance of qualified life and TLAA analyses 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

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(qualified life) is established for equipment within the scope of the program and an appropriate action such as replacement, refurbishment, or reanalysis is taken prior to or at the end of the equipment qualified life so that the aging limit is not exceeded. Changes to material activation energy values as part of a reanalysis are justified. The 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.

The Environmental Qualification of Electric Equipment aging management program will be enhanced to:

1. Add activities to visually inspect accessible, passive EQ equipment located in adverse localized environments at least once every 10 years. The first periodic visual inspection will be performed prior to the second period of extended operation.
2. Establish acceptance criteria for the visual inspections of accessible, passive EQ equipment located in adverse localized environments.

These enhancements will be implemented no later than six months prior to the second period of extended operation. New visual inspections of accessible, passive EQ equipment located in adverse localized environments will be completed no later than six months prior to the second period of extended operation, or no later than the last refueling outage prior to the second period of extended operation.

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R.4.0 TIME-LIMITED AGING ANALYSES

R.4.1 Identification and Evaluation of Time-Limited Aging Analyses

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of Time-Limited Aging Analyses (TLAAs) for the period of extended operation be provided. The TLAAs identified and evaluated to meet these requirements are described below.

10 CFR 54.21(c)(2) also requires that the application for a renewed license include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based upon TLAAs as defined in 10 CFR 54.3. It also requires an evaluation that justifies the continuation of these exemptions for the period of extended operation. No plant-specific exemptions granted pursuant to 10 CFR 50.12 were identified for PBAPS that are based upon a TLAA. Therefore, no further evaluation is required for plant-specific exemptions granted pursuant to 10 CFR 50.12.

R.4.2 Reactor Vessel and Internals Neutron Embrittlement Analyses

10 CFR 50.60 requires that all light-water reactors meet the fracture toughness, P-T limits, and material surveillance program requirements for the reactor coolant pressure boundary as set forth in 10 CFR 50, Appendices G and H. The current reactor pressure vessel embrittlement calculations for PBAPS that evaluate reduction of fracture toughness of the Units 2 and 3 reactor pressure vessel beltline materials for 60 years are based upon a predicted end-of-license fluence applicable for 54 Effective Full Power Years (EFPY). [The GNF3 fuel introduction revised fluence evaluation resulted in a reduction to 49.7 EFPY for Unit 2 and 47.5 EFPY for Unit 3 for the current P-T curves.](#) These analyses have been identified as TLAAs as defined in 10 CFR 54.21(c) and have been reevaluated for the increased neutron fluence associated with 80 years of operation as described in the subsections below. These subsections also include evaluations of the increased neutron fluence on reactor internal components, including potential loss of preload for the core plate rim hold-down bolts, jet pump slip joint clamps, jet pump auxiliary spring wedge assemblies, and jet pump riser clamps; as well as irradiation-enhanced stress relaxation of the replacement core plate extended life plugs.

R.4.2.1 Reactor Vessel and Internals Neutron Fluence Analyses

High energy (>1 MeV) neutron fluence has been projected for 80 years and 70 Effective Full Power Years (EFPY). The fluence projections have been used in the evaluation of the neutron embrittlement TLAAs. Two different methodologies have been used in developing the projections as described below.

- The NRC approved General Electric Hitachi (GEH) Discrete Ordinates Transfer (DORT) methodology has been used in developing the fluence projections and associated reactor vessel embrittlement analyses as documented in Sections R.4.2.1.1, and R.4.2.2 through R.4.2.7.

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- Transware Radiation Analysis Modeling Application (RAMA) methodology has been used to develop 80-year fluence projections for reactor vessel internal components that are used in evaluating reactor vessel internal component TLAAs in Sections R.4.2.1.2, and R.4.2.8 through R.4.2.12.

There has been no combination of the two methodologies applied to any component. Below is a summary of PBAPS historical operating power levels which have been considered in developing both 80-year fluence projections:

- The Original Licensed Thermal Power (OLTP) level for PBAPS Units 2 and 3 was 3293 MWt.
- By Amendment Nos. 198 and 211 (Units 2 and 3 respectively), the NRC approved an approximate 5.0 percent stretch power uprate to 3458 MWt in the mid-1990s.
- By Amendment Nos. 247 and 250 (Units 2 and 3, respectively) dated November 22, 2002, the NRC approved a 1.62 percent measurement uncertainty recapture (MUR) uprate that authorized an increase in the maximum thermal power level from 3458 MWt to 3514 MWt.
- By Amendment Nos. 293 and 296 dated August 25, 2014, the NRC approved a 12.4 percent EPU (Extended Power Uprate) that authorized an increase in the maximum thermal power level from 3514 MWt to the licensed thermal power level of 3951 MWt for both units.
- By Amendment Nos. 305 and 309, March 21, 2016 the NRC approved a Maximum Extended Load Line Limit Analysis Plus (MELLA+) operating strategy in 2016 in accordance with NEDC-33006P-A for both units.
- By Amendment Nos. 316 and 319 dated November 15, 2017, the NRC approved a 1.66 percent Measurement Uncertainty Recapture (MUR) power uprate to 4016 MWt at mid-Cycle 22 for Unit 2 and at the beginning of Cycle 22 for Unit 3.

The current uprated power level of 4016 megawatts thermal (MWt) is the maximum power level evaluated for the second period of extended operation.

R.4.2.1.1 Reactor Vessel Neutron Fluence Analyses

Reactor vessel fluence projections for 80 years and 70 EFPY have been developed using the NRC approved General Electric Hitachi (GEH) Discrete Ordinate Transfer (DORT) methodology. The GEH methodology adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation. The 70 EFPY fluence projection values have been used in the evaluation of the neutron embrittlement TLAAs for reactor vessel beltline materials, which include the reactor vessel plate materials, welds, and nozzle forgings. Fluence projections have been developed to evaluate fluence-based reactor vessel

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TLAAs and to determine when specified fluence threshold values may be exceeded that are used to invoke specific aging management requirements, such as inspections, for these components.

These 70 EFPY fluence projections will be validated by the Neutron Fluence Monitoring (R.3.1.2) aging management program during the second period of extended operation.

The Unit 2 and Unit 3 RPV beltline component fluence analyses have been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.2.1.2 Reactor Vessel Internals Neutron Fluence Analyses

Fluence projections have been developed for 80 years and 70 EFPY for reactor vessel internal components using the Transware Radiation Analysis Modeling Application (RAMA) Fluence Methodology. Use of this model was performed in accordance with NRC Regulatory Guide 1.190. The fluence projections have been developed for specific reactor vessel internal components to evaluate fluence-based TLAAs and to determine when specified fluence threshold values may be exceeded that are used to invoke specific aging management requirements, such as inspections, for these components.

The Unit 2 and Unit 3 reactor vessel internal component fluence analyses have been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.2.1.3 RPV Supplemental Metallurgical Sensitivity Evaluation

The RPV extended beltline elevations are the upper and lower elevations bounding the region of the RPV vessel where fluence values are projected to equal or exceed $1.0E+17$ n/cm².

NUREG-2192 and 10 CFR 50.60 recommend fracture toughness evaluations of RPV components which are projected to experience fluence values of $1.0E+17$ n/cm² or more. RPV fluence and upper extended beltline elevation projections for 80 years at 70 EFPY were performed using the GEH DORT methodology as described in Section R.4.2.1.1. Fracture toughness evaluations of RPV components within the extended beltline elevations are described in Section R.4.2.2 through R.4.2.7.

The Neutron Fluence Monitoring aging management (R.3.1.2) program is an existing condition monitoring program that monitors and tracks neutron fluence to RPV components to ensure that fluence projections described on Section R.4.2.1.1 remain valid through second period of extended operation.

Sensitivity evaluations have been performed to address a recent NRC concern that fluence projections above the active fuel region may have greater uncertainty than fluence projections for RPV elevations within the active fuel region. The concern is that water density sensitivity analyses described in Section 7.1, "Calculation Uncertainties," of NEDC-32983P-A, Revision 2, are not applicable to the above core water density distribution because the

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generically approved calculational model does not extend above the active fuel region based on Figure 2-2 of in NEDC-32983P-A, Revision 2. Also, projected fluence results for RPV elevations above the active fuel have not been benchmarked. Non-conservative fluence projections could potentially result in a non-conservatively low upper extended beltline elevation and exclude the metallurgically evaluation of portions of the RPV that will actually be exposed to $1.0E+17$ n/cm² or more, prior to the end of the period of extended operation.

Because of this uncertainty, work is ongoing with the ANS 19.10 standards committee to revise the guidance within RG 1.190 to more explicitly outline its expectations for fluence projections within the upper plenum. In addition, the sensitivity evaluations discussed below have been performed to provide reasonable assurance that all components that may experience fluence values of $1.0E+17$ n/cm² or more, meet established metallurgical acceptance criteria.

As shown on Figure 4.2.1-1, for Unit 2, the calculated 70 EFPY upper extended beltline elevation ^{Security-Related Information Withheld under 10 CFR 2.390} resulted in the successful metallurgical evaluation of: all the lower-intermediate vessel wall plates, all the lower vessel wall plates, all the V1A/B/C and all V2A/B/C axial welds, the H2 circumferential weld, and the N16 nozzles, as described in Sections R.4.2.2 through R.4.2.7.

The Unit 2 intermediate plates, the H3 circumferential weld, and V3 axial welds ^{Security-Related Information Withheld under 10 CFR 2.390} and have not been metallurgically evaluated in Sections R.4.2.2 through R.4.2.7. Although it is not reasonable that the uncertainty associated with the NRC's concern would result in the extension of the upper extended beltline elevation ^{Security-Related Information Withheld under 10 CFR 2.390} at 70 EFPY, a supplemental metallurgical sensitivity evaluation and justification was performed on all the Unit 2 intermediate plates, the H3 circumferential weld, and all V3A/B/C axial welds. Results of the supplemental metallurgical sensitivity evaluation concluded that, even with a postulated fluence value that is more than 20 times greater than expected, these reactor vessel components meet all the established metallurgical acceptance criteria for ART, USE/EMA, circumferential weld inspection relief, axial weld probability, and reflood thermal shock.

As shown on Figure 4.2.1-2, the Unit 3 lower-intermediate plates, are ^{Security-Related Information} shorter than the Unit 2 lower-intermediate plates. As a result, the Unit 3 projected upper extended beltline ^{Security-Related Information Withheld under 10 CFR 2.390} extends onto portions of the intermediate plates and V3A/B/C welds and envelops the Unit 3 H3 weld, while for Unit 2 the upper extended beltline elevation does not reach these components. Therefore, for Unit 3, the calculated 70 EFPY upper extended beltline elevation resulted in the successful metallurgical evaluation of: all the intermediate vessel wall plates, all the lower-intermediate vessel wall plates, all the lower vessel wall plates, all the V1A/B/C, V2A/B/C, and V3A/B/C axial welds, the H2 and H3 circumferential weld, and the N16 nozzles, as described in Sections R.4.2.2 through R.4.2.7.

Unlike the Unit 2 intermediate plates, V3A/B/C welds, and H3 weld, which are located above the Unit 2 upper extended beltline elevation ^{Security-Related Information Withheld} and

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therefore excluded from fracture toughness evaluation described in Sections R.4.2.2 and through R.4.2.7, the Unit 3 intermediate plates and V3A/B/C welds Security-Related Information Withheld under 10 CFR 2.390, and H3 weld Security-Related Information Withheld under 10 CFR 2

are evaluated for fracture toughness as described in Sections R.4.2.2 and through R.4.2.7. Nevertheless, because of the uncertainty associated with the NRC's concern, a supplemental metallurgical sensitivity evaluation and justification was performed on all the Unit 3 intermediate plates and associated welds. Results of the supplemental metallurgical sensitivity evaluation concluded that, even with a postulated fluence value that is more than 20 times greater than expected, these reactor vessel components meet all the established metallurgical acceptance criteria for ART, USE/EMA, circumferential weld inspection relief, axial weld probability, and reflood thermal shock.

The supplemental metallurgical sensitivity evaluation does not introduce additional Unit 2 and 3 RPV components, that require fracture toughness evaluation per the recommendations in NUREG-2192 and 10 CFR 50.60. The supplemental metallurgical sensitivity evaluation serves as an analysis and justification that provides reasonable confidence that RPV components Security-Related would meet the established metallurgical acceptance criteria should future industry resolution of the concern result in the extension of the upper extended beltline elevation Security-Related Information Withheld under 10 CFR 2.390.

The Unit 2 and Unit 3 N9 nozzles are the lowest RPV components above the upper extended beltline elevation that have not been metallurgically evaluated in section R.4.2.2 through R.4.2.7 or in the supplemental metallurgical sensitivity evaluation. These are single 4-inch Control Rod Drive Hydraulic System Return nozzles Security-Related Information Withheld under 10 CFR 2.390

It is not credible that the uncertainty associated with the NRC's concern would result in an extension of the upper extended beltline elevation Security-Related Information Withheld under 10 CFR 2.390

This NRC concern has been entered in to the PBAPS corrective action program. Results from industry activities associated with the ANS 19.10 standards committee will be incorporated into the Neutron Fluence Monitoring aging management (R.3.1.2) program as appropriate.

Security-Related Information Withheld under 10 CFR 2.390

R.4.2.2 Reactor Vessel Upper-Shelf Energy (USE) Analyses

Appendix G of 10 CFR 50, Paragraph IV.A.1.a, states that reactor vessel beltline materials must have Charpy upper-shelf energy (USE) throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

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The PBAPS Units 2 and 3 reactor vessels were designed and fabricated prior to the current requirements, and as a result, there is insufficient data available to establish the initial unirradiated USE value for all beltline materials for these reactor vessels. Therefore, the current licensing basis Charpy USE evaluations are based upon Equivalent Margin Analysis (EMA) as specified in BWRVIP-74-A, which meets the alternative requirements specified above. The PBAPS Units 2 and 3 February 2017 MUR amendment submittal reevaluated EMA for 60 years and 54 EFPY fluence values. Therefore, these analyses have been identified as TLAAs requiring evaluation for the second period of extended operation.

An EMA has been performed for the limiting beltline plate and weld materials for 80 years of operation at 70 EFPY, and then compared against the 54 EFPY limits defined in Appendix B of BWRVIP-74-A. The comparison concluded that the Units 2 and 3 vessel materials meet the 54 EFPY limits defined in Appendix B of BWRVIP-74-A and the USE values for Unit 2 and Unit 3 reactor vessel beltline materials have been satisfactorily evaluated for the second period of extended operation based upon the updated EMA values determined using 80-year (70 EFPY) fluence projections. Therefore, the Equivalent Margins Analyses have been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.2.3 Reactor Vessel Adjusted Reference Temperature (ART) Analyses

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the P-T limit curves to account for irradiation effects. The initial nil-ductility reference temperature (RT_{NDT}) is the temperature at which an unirradiated ferritic steel material changes in fracture characteristics from ductile to brittle behavior. RT_{NDT} is evaluated according to the procedures in the ASME Code. Neutron embrittlement increases the RT_{NDT} beyond its initial value.

10 CFR 50, Appendix G, defines the fracture toughness requirements for the life of the vessel. The shift in the initial RT_{NDT} (ΔRT_{NDT}) is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) determines how much higher the vessel temperature must be raised for the material to continue to act in a ductile manner. The ART is defined as: Initial RT_{NDT} + ΔRT_{NDT} + Margin. The PBAPS Units 2 and 3 February 2017 MUR amendment submittal reevaluated the adjusted reference for 60 years and 54 EFPY fluence values. Therefore, these analyses have been identified as TLAAs requiring evaluation for the second period of extended operation.

70 EFPY ART values have been determined for PBAPS Units 2 and 3 beltline materials using the methodology specified in Regulatory Guide 1.99, Revision 2. The 70 EFPY ART values of the limiting beltline materials for each unit remain below 200 degrees F, which is the RT_{NDT} limit.

The 70 EFPY ART analyses have been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(ii).

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R.4.2.4 Reactor Vessel Pressure-Temperature (P-T) Limits

10 CFR 50 Appendix G requires that the reactor pressure vessel be maintained within established pressure-temperature (P-T) limits, including heatup and cooldown operations. These limits specify the minimum acceptable reactor coolant temperature as a function of reactor pressure. As the reactor pressure vessel is exposed to increased neutron irradiation over time, its fracture toughness is reduced. The P-T limits must account for the change in material properties due to anticipated reactor vessel fluence.

The current Pressure-Temperature limit curves are located in a Pressure-Temperature Limits Report and are based upon 53 EFPY fluence projections that were considered to represent the amount of power to be generated over 60 years of plant operation under MUR conditions. The currently licensed P-T curves were developed for up to 54 EFPY at the EPU power level of 3951 MWt. The 54 EFPY MUR analyses resulted in a minor increase in adjusted reference temperature (ART) of less than 0.5 degrees F at a few beltline locations. These minor increases slightly exceed a few of the ART values assumed for the currently licensed P-T curves. Therefore, currently licensed P-T curves have been evaluated for MUR conditions and concluded to be valid for up to 53 EFPY at the MUR power level of 4016 MWt. [The GNF3 fuel introduction revised fluence evaluation resulted in a reduction to 49.7 EFPY for Unit 2 and 47.5 EFPY for Unit 3 to maintain the applicability of the current P-T curves.](#) The P-T limits have been identified as TLAA's requiring evaluation for the second period of extended operation.

In accordance with NUREG-2192, Section 4.2.2.1.4, the P-T limits for the second period of extended operation will be updated at the appropriate time through the plant's Administrative Section of the PBAPS Technical Specification 5.6.7, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" and the plant's PTLR process. This process will ensure that the P-T limits for the second period of extended operation will be updated prior to expiration of the 53 EFPY associated with the P-T limit curves.

Therefore, the effects of aging on the intended function(s) of the reactor vessels will be adequately managed through the second period of extended operation, as described in Technical Specification 5.6.7, in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.2.5 Reactor Vessel Circumferential Weld Failure Probability Analyses

PBAPS has previously applied for and been granted relief from RPV circumferential weld inspection for the Units 2 and 3 vessels. The relief from inspection is based on assessment of the probability of failure of the limiting circumferential weld. This assessment is based on 54 EFPY fluence values associated with 60 years of operation, and has therefore been identified as a TLAA requiring evaluation for the second period of extended operation.

In order to evaluate the PBAPS Units 2 and 3 circumferential weld failure probability for 80 years, 70 EFPY fluence values have been projected for each

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circumferential weld and mean RT_{NDT} values have been then determined. The mean RT_{NDT} values have been compared to circumferential weld probability analysis at 64 EPFY specified in the NRC's Final Safety Evaluation Report (FSER) of BWRVIP-05. The PBAPS mean RT_{NDT} values based on 70 EPFY are significantly less than the NRC RT_{NDT} values based on 64 EPFY used in determining the conditional failure probability in NRC's FSER of BWRVIP-05 dated July 28, 1998. Therefore, the NRC conditional failure probability is bounding for PBAPS Units 2 and 3, consistent with the requirements defined in GL 98-05.

Reapplication for relief from circumferential weld examination will be made in accordance with 10 CFR 50.55a(a)(3) in time for NRC review and approval prior to the second period of extended operation. The plant-specific information described above demonstrates that at the end of the second period of extended operation, the circumferential beltline weld materials meet the limiting conditional failure probability for circumferential welds specified in the NRC's FSER of BWRVIP-05. These analyses will be managed in accordance with 10 CFR 54.21(c)(1)(iii) by requesting relief from circumferential weld inspection using the 10 CFR 50.55a process.

R.4.2.6 Reactor Vessel Axial Weld Failure Probability Analyses

The BWRVIP recommendations for inspection of reactor pressure vessel shell welds in BWRVIP-05 include examination of 100 percent of the axial welds and inspection of the circumferential welds only at the intersections of these welds with the axial welds. The Staff provided separate conditional failure probability assessments in the Supplement to the Final Safety Evaluation of the BWRVIP-05 Report, dated March 7, 2000, and calculated a RPV failure frequency of $5.02E-06$ due to failure of limiting axial welds in the BWR fleet. Since these NRC Staff failure probability assessments are applicable to PBAPS Units 2 and 3, they are identified as TLAA's requiring evaluation through the second period of extended operation.

In order to evaluate axial weld failure probability analyses for 80 years for the PBAPS Units 2 and Unit 3 vessels, 70 EPFY fluence projections have been developed for the inside surface of the limiting axial welds. Using the bounding inside surface fluence value, the mean RT_{NDT} values have been determined for these welds, where the mean RT_{NDT} value does not include the margin term (M) described in RG 1.99, Revision 2, consistent with the evaluation methodology described in Section 2.1 of the March 7, 2000 supplement to the final safety evaluation.

The limiting PBAPS Units 2 and 3 axial weld mean RT_{NDT} values at 70 EPFY (without margin) are bounded by the mean RT_{NDT} values (without margin) determined for the limiting reactor in the March 7, 2000 supplement to the final safety evaluation. Therefore, the NRC's calculated RPV failure frequency of $5.02E-06$ due to failure of limiting axial welds in the BWR fleet, is bounding for PBAPS Units 2 and 3. Therefore, this analysis has been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

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R.4.2.7 Reactor Vessel Reflood Thermal Shock Analysis

10CFR50 Appendix A, General Design Criterion 31 requires that the reactor coolant pressure boundary of a light water reactor be designed such that it possesses adequate margin against non-ductile failure for all postulated conditions. For boiling water reactors, this requirement is demonstrated both by development of Pressure-Temperature Limit Curves, which are addressed in Section R.4.2.4, and by reference to a generic fracture mechanics analysis that evaluates the effects of the limiting Loss of Coolant Accident (LOCA) event.

The generic fracture mechanics analysis evaluates the effects of a postulated LOCA on the structural integrity of a reactor pressure vessel. The rupture of a main steam line was determined to bound all other LOCA events with respect to this evaluation. After the rupture, several emergency core cooling systems are activated at different times and the vessel is flooded with cooling water. The vessel depressurization and the subsequent injection of cold water to reflood the reactor vessel produce a rapid reduction in temperature and high thermal stresses in the vessel. The analysis concludes that the reactor pressure vessel has a considerable margin to failure by brittle fracture even in the presence of postulated pre-existing flaws. This generic analysis envelopes PBAPS and is based on BWR vessel material properties and cumulative fluence assumed for 40 years of operation. The PBAPS Units 2 and 3 February 2017 MUR amendment submittal reevaluated the RPV reflood thermal shock analysis for 60 years and 54 EFPY fluence values. Therefore, this analysis has been identified as a TLAA requiring evaluation for the second period of extended operation.

An updated 80-year fracture mechanics evaluation was performed for the reflood thermal shock event to evaluate components with the limiting material properties from the PBAPS Units 2 and 3 RPV beltline plates, axial welds, and circumferential welds, which bounds the remainder. The analysis used projected 70 EFPY fluence values and determined that during the second period of extended operation, each RPV has sufficient toughness margin to prevent unacceptable flaw propagation due to thermal shock during reflooding after LOCA events.

The reactor pressure vessel reflood thermal shock analysis has been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.2.8 Core Shroud Reflood Thermal Shock Analysis

Neutron irradiation embrittlement may affect the ability of reactor vessel core shroud to withstand a low-pressure coolant injection thermal shock transient. The PBAPS reactor vessel core shrouds, which were fabricated from Type 304 stainless steel, have been analyzed for a low-pressure coolant injection reflood thermal shock transient considering the embrittlement effects of neutron irradiation exposure as documented in UFSAR Section 3.3.5.4. The core shrouds receive the maximum irradiation on the inside surface approximately opposite the midpoint of the fuel centerline. The maximum thermal shock

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stress in this region was determined to be 155,700 psi at the midpoint of the shroud, which is equivalent to 0.57 percent strain during the reflood thermal shock transient. This analysis has been identified as a TLAA requiring evaluation for the second of extended operation.

This issue was previously evaluated as a TLAA for the 60-year license renewal of the Brown's Ferry plants. Similar to PBAPS, the Brown's Ferry maximum thermal shock stress in the shroud was determined to be 155,700 psi with an equivalent 0.57 percent strain. In associated RAI responses, Browns Ferry provided material destructive testing results of highly irradiated Type 304 stainless steel to demonstrate that the core shrouds would withstand a low-pressure coolant injection event. These conclusions were accepted by the Staff. These same material destructive testing results can be used to disposition neutron irradiation embrittlement of the PBAPS core shrouds and their ability to withstand a low-pressure coolant injection thermal shock transient. For this TLAA, Brown's Ferry evaluated two material properties: "reduction in area" and "elongation".

For the "reduction in area" material property the Brown's Ferry evaluation concluded that a fluence value of $5.34\text{E}+21$ n/cm² over the life of the core shroud results in sufficient ductility during the reflood thermal shock transient to resist the 155,700 psi stress. For the "elongation" material property the Brown's Ferry evaluation concluded that a fluence value of $8.0\text{E}+21$ n/cm² over the life of the core shroud results in strain values, during the reflood thermal shock transient, which bound 0.57 percent strain. The maximum 80-year (70 EFPY) shroud weld fluence values for PBAPS Unit 2 is $3.63\text{E}+21$ n/cm² and for Unit 3 is $3.45\text{E}+21$ n/cm². Therefore, the conclusion for the Brown's Ferry core shrouds analysis also applies to the PBAPS core shrouds.

The projected 80-year (70 EFPY) shroud fluence values for PBAPS Units 2 and 3 are less than assumed fluence values established in the Brown's Ferry evaluation. Therefore, the conclusion for the Brown's Ferry shrouds also applies to the PBAPS shrouds and the analysis remains valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.2.9 Core Plate Rim Hold-Down Bolt Loss of Preload Analysis

The RPV core plate is attached to the core support structure by 34 stainless steel hold-down bolts arranged along the rim of the plate. These bolts are subject to stress relaxation (loss of preload) due to irradiation effects. An analysis was performed concluding that a reduction in preload as high as 19 percent over the 40-year life of the bolts is acceptable to meet design requirements. A subsequent reevaluation determined that this maximum relaxation value of 19 percent is applicable to an average fluence value of $8.0\text{E}+19$ n/cm² over the entire length of the bolt located at the azimuthal location with peak fluence. These analyses were identified as TLAA's.

In order to determine if these analyses will remain valid for 80 years, RAMA fluence projections were prepared for 70 EFPY for the core plate rim hold-down bolts located at the azimuthal location with peak fluence. The bolts at the peak azimuthal location with the highest fluence for PBAPS Unit 2 and Unit 3 are bolt

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numbers 4, 15, 21, and 32, which have the same fluence values due to core symmetry. Fluence projections were made at the centerline of 27 discrete points equally spaced along the tensioned portion of these limiting bolts. For Unit 2, the 70 EFPY average fluence value of the limiting bolts is $6.57\text{E}+19$ n/cm². For Unit 3, the 70 EFPY average fluence value of the limiting bolts is $6.53\text{E}+19$ n/cm². Since these average fluence values are less than the $8.0\text{E}+19$ n/cm² average fluence value previously evaluated in the TLAA, which resulted in an acceptable maximum relaxation value of 19 percent, the TLAA remains bounding for 70 EFPY and 80 years of operation. Therefore, the analysis remains valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.2.10 Jet Pump Slip Joint Repair Clamp Loss of Preload Analysis

Jet pump slip joint repair clamps have been designed and installed on jet pumps in PBAPS Unit 2 to minimize slip joint vibration and wear of the jet pump assemblies. Eight clamps were installed in 2004, one was installed in 2008, and two were installed in 2014. None have been installed in Unit 3. The clamps apply a lateral preload to the slip joint, between the exit end of the inlet-mixer and the entrance end of the diffuser, to dampen jet pump vibration. The design analysis for the clamps evaluated a neutron fluence value of $1.115\text{E}+20$ n/cm² for a 40-year design life of the clamp, and demonstrated that loss of preload resulting from neutron fluence during the design life of the clamps was acceptable. This analysis was identified as a TLAA.

In order to evaluate this TLAA, fluence at clamp locations inside the reactor vessel was determined from initial clamp installation in 2004 through the end of the second period of extended operation. The peak fluence value is projected to reach of $1.17\text{E}+19$ n/cm² at the end of the second period of extended operation. Therefore, since this value is less than the design fluence value of $1.115\text{E}+20$ n/cm², this TLAA has been determined to remain valid through the second period of extended operation. Therefore, the design analysis remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.2.11 Jet Pump Auxiliary Spring Wedge Assembly Loss of Preload Analysis

The PBAPS jet pump (JP) assemblies have had auxiliary spring wedge assemblies installed to maintain lateral support for the jet pump inlet mixer. The design stress analysis considered potential aging effects, including loss of preload due to radiation effects based upon a design life of 40 years. The auxiliary spring wedge assemblies were installed in Unit 2 on JP 10, 12, and 18 in 2004, installed on JP 14, 19, and 20 in 2006, replaced on JP 10 and JP 18 in 2014, and removed from JP 20 in 2014. Therefore, the earliest installed auxiliary spring wedge assembly will have a maximum in service time of approximately 49 years by the end of the second period of extended operation in August 2053.

An auxiliary spring wedge assembly was installed in Unit 3 on JP 14 in 2001 and will have a maximum in service time of approximately 53 years by the end of the second period of extended operation in July of 2054. Also, an auxiliary

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spring wedge assembly was installed in Unit 3 on JP 09 in 2001 and removed during the 2017 refueling outage.

The auxiliary spring wedge assembly design analysis determined that the fluence levels in the regions where the auxiliary wedges are installed on the jet pumps are less than $5.0E+20$ n/cm² for a 40-year design life. An assumed 10 percent load relaxation was used in the analysis to account for loss of preload due to thermal and radiation effects. This analysis was identified as a TLAA.

In order to evaluate this TLAA, fluence projections were calculated for the limiting auxiliary spring wedge assembly for each unit. For Unit 2, the maximum fluence at the limiting auxiliary spring wedge assembly was determined to be $1.53E+20$ n/cm² for the first installed auxiliary spring wedge assembly installed in 2004 to the end of the second period of extended operation. For Unit 3, the maximum fluence at the limiting auxiliary spring wedge assembly was determined to be $8.12E+18$ n/cm² for the first installed auxiliary spring wedge assembly installed in 2001 to the end of the second period of extended operation. Each of these 70 EFPY fluence values are less than the $5.0E+20$ n/cm² fluence value assumed in the design structural analysis for which the 10 percent loss of preload allowance was applied. Therefore, the design analysis remains valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.2.12 Jet Pump Riser Repair Clamp Loss of Preload Analysis

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 for jet pump numbers 01/02 and 13/14. A mechanical clamping system designed to structurally replace these welds was installed in 1998. Since these clamshell-style clamps use bolts to maintain the proper clamping force, loss of preload due to neutron irradiation stress relaxation was a design consideration. The design specification assumed that the neutron fluence at the riser pipe clamp location would not exceed $2.5E+19$ n/cm² for the service life of the clamps, until the end of the initial 40 years of operation. Since the neutron fluence value was based on the initial 40 years of operation, this analysis has been identified as a TLAA.

In order to determine if this fluence assumption will remain valid through 80 years of operation, neutron fluence was projected through the second period of extended operation. The 70 EFPY fluence value at the limiting Unit 3 jet pump riser clamp location was determined to be $5.29E+15$ n/cm², which is less than the $2.5E+19$ n/cm² fluence value assumed in the design specification for the clamp. Therefore, the design analysis remains valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.2.13 Replacement Core Plate Extended Life Plug Irradiation-Enhanced Stress Relaxation Analysis

The original design of the PBAPS Units 2 and 3 core plates included holes for bypass flow. It was discovered that the flow through the holes produced high velocity jets that impinged on the in-core instrument tubes, which subjected

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them to high levels of flow induced vibration, which led to wear on the adjacent fuel channels. The core plate holes were plugged to prevent the unwanted flow induced vibration. During the PBAPS Unit 3 fall 2001 refueling outage, all 129 reactor core plate plugs were replaced with extended life core support plugs. During the PBAPS Unit 2 spring 2012 refueling outage, all 129 reactor core plate plugs were replaced with extended life plugs. The extended life core support plugs have a service life of 35 EFPY corresponding to a fluence of $5.25E+20$ n/cm². Due to the effects of irradiation-enhanced stress relaxation, the amount of force applied by the plug mandrel spring is dependent on the accumulated neutron fluence. The plug mandrel spring essentially holds the extended life core support plug tight in the core plate, against a force of 46.7 pounds created by the differential pressure in the core plate. Therefore, irradiation-enhanced stress relaxation of the mandrel spring due to neutron irradiation has been identified as a TLAA and was evaluated through the second period of extended operation.

Reevaluation of the service life of the extended life core support plugs concluded that the service life can be extended by an additional 20 EFPY for total service life of 55 EFPY. The reevaluation concluded that at the end of the 55 EFPY period the core support plug mandrel springs are estimated to experience fluence values of $8.25E+20$ n/cm² with a resulting end of life mandrel spring preload of 111 pounds which exceeds the force acting on the plug due to the differential pressure, by margin of at least 100 percent. Therefore, the extended life core support plugs would not reach the end of their service life by 2067 for Unit 2 and 2056 for Unit 3, assuming an operating capacity of 100 percent. Since the second period of extended operation will end in 2053 for Unit 2 and 2054 for Unit 3, the evaluated service life of 55 years will not be exceeded during the second period of extended operation.

The analysis has been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.2.14 First License Renewal Application Core Shroud IASCC and Embrittlement Analysis

Section 4.3.2.2 of the first PBAPS License Renewal Application (LRA) presents a fluence threshold value of $5.0E+20$ n/cm² beyond which IASCC and embrittlement may occur in BWR vessel internal components. Section 4.3.2.2 of the first PBAPS LRA concludes that the expected fluence on the inner surfaces of the core shroud would be $4.5E+20$ n/cm² at the end of the 60-year first period of extended operation. Therefore, supplemental aging management of the core shroud to address IASCC and embrittlement is not required through the first period of extended operation. Since the core shroud analysis presented in Section 4.3.2.2 of the first PBAPS LRA determined that IASCC and embrittlement were not applicable aging effects expected to occur in 60 years, this analysis has been identified as a TLAA that requires evaluation for the second period of extended operation.

Fluence values for the PBAPS Unit 2 and Unit 3 core shroud are projected to exceed the threshold of $5.0E+20$ n/cm² before the end of the second period of extended operation. Therefore, the core shroud will be inspected periodically

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for cracking and loss of fracture toughness (embrittlement) during the second period of extended operation in accordance with the BWR Vessel Internals (R.2.1.7) program.

The effects of aging on the intended function(s) of the reactor vessel core shroud will be adequately managed through the second period of extended operation by the BWR Vessel Internals (R.2.1.7) program, in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.2.15 Unit 3 Core Spray Replacement Piping Bolting Loss of Preload Evaluation

In 2013, portions of the Unit 3 Core Spray System piping segment located inside the reactor vessel were replaced from the thermal sleeves in reactor pressure vessel nozzles N5A and N5B to the shroud wall. A design report associated with this replacement evaluated loss of preload due to fluence on the most limiting bolting component. The loss of preload evaluation assumed a fluence value of $3.6E+19$ n/cm² over a 40-year service life. This resulted in a reduction of preload to 3,775 pounds at the end of the 40-year period, which exceeds and meets the acceptance criterion of 3,504 pounds. Since the design report evaluated the effects of fluence on loss of preload over a 40-year service life, until 2053, the evaluation was identified as a TLAA that requires evaluation for the second period of extended operation, which ends in 2054 for PBAPS Unit 3.

The associated loss of preload evaluation was reevaluated by GEH for an additional five years of service life for a total of 45 years, until 2058. The reevaluation applied a fluence value of $4.1E+19$ n/cm² which was assumed over 45 years. This resulted in a reduction of preload on the most limiting bolting component to 3,682 pounds through 2058, which exceeds and meets the acceptance criterion of 3,504 pounds. The Unit 3 replacement core spray piping bolting loss of preload evaluation has been satisfactorily projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.3 Metal Fatigue Analyses

Metal fatigue was considered explicitly in the design process for pressure boundary components designed in accordance with ASME Section III, Class 1 requirements. Metal fatigue was evaluated implicitly for components designed in accordance with ASME Section III, Class 2 or 3 requirements or ANSI B31.1 requirements. Each of these fatigue analyses and evaluations are considered to be Time-Limited Aging Analyses (TLAAs) requiring evaluation for the second period of extended operation in accordance with 10 CFR 54.21(c) as described below.

R.4.3.1 Transient Cycle and Cumulative Usage Projections for 80 years

Fatigue analyses are based upon explicit numbers and amplitudes of thermal and pressure transients usually described in design specifications. The intent of the design basis transient definitions is to bound a wide range of possible events with varying ranges of severity in temperature and pressure. Since the existing fatigue analyses are based upon a number of transient cycles

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postulated to bound 60 years of service, projection of the transient cycles through the second period of extended operation is required to demonstrate that the analyses and waivers remain valid.

Projections of the transient cycles' through the second period of extended operations were developed for 80 years and used as input to calculate projected 80-year Cumulative Usage Factor (CUF) and Environmentally Assisted Cumulative Usage Factor (CUF_{en}) values to determine whether the existing analyses remain valid for 80 years. The number of transient cycles, CUF values, and CUF_{en} values have been projected through the second period of extended operation. The following fatigue TLAs have been dispositioned using the projected number of transient cycles, CUF values, and CUF_{en} values through the second period of extended operation.

R.4.3.2 ASME Section III, Class 1 Fatigue Analyses

The PBAPS reactor pressure vessels (RPVs) were originally designed for 40 years of service in accordance with the ASME Code Section III, its interpretations, and applicable requirements, (including 1965 Winter Addendum for Unit 2 and 3) for Class 1 design requirements. The RPV Class 1 fatigue analyses determined the effects of transient cyclic loadings resulting from changes in system temperature and pressure and for seismic loading cycles. The fatigue analyses evaluated explicit numbers and types of transients that were postulated for the 40-year design life of the plant in the design specifications. These Class 1 explicit fatigue analyses were required to demonstrate that the CUF for each component will not exceed the design limit of 1.0 for all the postulated transients. As stipulated in PBAPS's first License Renewal Application (LRA) the original 40-year RPV explicit fatigue analyses were updated with 60-year projected transient cycle numbers. The 60-year evaluations now serve as the current licensing basis (CLB), and have been identified as TLAs for the second period of extended operation.

All PBAPS Class 1 piping systems were originally designed and evaluated in accordance with USAS (ANSI) B31.1 design requirements, which did not include explicit fatigue analysis. In the 1980s, PBAPS replaced reactor recirculation and the residual heat removal (RHR) system piping on both units. This replaced piping was designed in accordance with ASME Section III, 1980 Edition, including winter addenda through 1981, as Class 1 piping. In addition, the Unit 3, the flued-head penetrations for the RHR system were also analyzed for fatigue in accordance with ASME Section III, Class 1 requirements. Therefore, this replaced piping and flued-head penetrations were explicitly evaluated for fatigue. The remaining Unit 3 penetrations and all Unit 2 penetrations are designed in accordance with ASME Section III, Class 2 requirements, which do not include explicit fatigue analyses.

In the first PBAPS License Renewal Application all ASME Section III, Class 1 fatigue analyses were identified as TLAs, evaluated for 60 years of operation, and dispositioned in accordance with 10 CFR 54.21(c)(iii) for management of the aging effect using the current Fatigue Monitoring program. These Class 1 explicit fatigue analyses have been identified as TLAs that require evaluation for the second period of extended operation.

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Eighty-year CUF and CUF_{en} projections show that all ASME Section III, Class 1 fatigue analysis will continue to meet the ASME Section III design limit of 1.0 through the second period of extended operation. To ensure the projected CUF and CUF_{en} values remain acceptable for the 80-year period of operation, the Fatigue Monitoring (R.3.1.1) program will monitor cumulative CUF and CUF_{en} for limiting locations through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.3.3 ASME Section III, Class 1 Fatigue Waivers

The PBAPS reactor pressure vessels (RPVs) were originally designed for 40 years of service in accordance with the ASME Code Section III, its interpretations, and applicable requirements, (including 1965 Winter Addendum) for Class 1 design requirements. The design stress reports for the Unit 2 and Unit 3 RPVs include fatigue waivers that determined that some RPV nozzles did not require explicit fatigue analyses because the criteria from ASME Section III, Paragraph N-415.1 were satisfied. The PBAPS RPV nozzles with fatigue waivers are the: Steam Outlet Nozzles, Liquid Control Nozzles, Instrumentation Nozzles, Vent Nozzles, Jet Pump Instrument Nozzles, and the RPV Drain Nozzles.

Since, the ASME Section III, Paragraph N-415.1 fatigue waiver criteria require postulated cycle input for the intended operating life of the plant, these fatigue waivers are TLAA's. Therefore, fatigue waiver evaluations were reevaluated for the second period of extended operation using the 80-year projected number of transients. The results of the reevaluation show that the criteria in ASME Section III, Paragraph N-415-1 remain satisfied through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.3.4 ASME Section III, Class 2 and 3 and ANSI B31.1 Allowable Stress Analyses

Piping designed in accordance with ASME Section III, Class 2 or 3, or ANSI B31.1 Piping Code design rules is not required to have an explicit analysis of cumulative fatigue usage, but cyclic loading is considered in the design process. If the numbers of anticipated thermal cycles exceed specified limits, these codes require the application of a stress range reduction factor to the allowable stress to prevent damage from cyclic loading. This is an implicit fatigue analysis since it is based upon the anticipated number of cycles for the life of the piping system.

These codes first require the overall number of thermal and pressure cycles expected during the plant lifetime of these components to be determined. A stress range reduction factor is then determined for that number of cycles using the applicable design code. If the total number of cycles is 7,000 or less, the stress range reduction factor of 1.0 is applied, which would not reduce the allowable stress values. For higher numbers of cycles, the stress range reduction factors limit the allowable stresses that can be applied to the piping.

Portions of the following Class 2 and 3 and ANSI B31.1 piping systems within the scope of license renewal are directly connected to Reactor Coolant System (RCS) and are affected by the same operational transients that result in

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thermal cycles for the attached Class 1 RCS piping: Control Rod Drive, Core Spray, Feedwater, Main Steam, Offgas and Recombiner, Primary Containment Isolation, Reactor Pressure Vessel and Internals, Reactor Recirculation, Reactor Vessel Instrumentation, Residual Heat Removal, and Standby Liquid Control Systems. These transient cycles have been projected for 80 years. The projections demonstrate that the total number of thermal cycles for these piping systems will not exceed 50 percent of the 7,000-cycle threshold that would result in a reduction in the stress range reduction factor. Therefore, these TLAAAs have been demonstrated to remain valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

For the remaining Class 2 and 3 and ANSI B31.1 piping systems within the scope of license renewal that are affected by thermal and pressure transients that are different than the RCS transients, an operational review was performed. These piping systems include portions of the Auxiliary Steam, Emergency Diesel Generator, Fire Protection, High Pressure Coolant Injection, Process Sampling, Reactor Core Isolation Cooling, and Reactor Water Cleanup Systems.

The review concluded that the total number of thermal cycles for these systems, projected through the period of extended operation, will not exceed 68 percent of the 7,000-cycle threshold. Therefore, the stress range reduction factors originally selected for the Class 2 and 3 and ANSI B31.1 piping systems remain applicable and these TLAAAs have been demonstrated to remain valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.3.5 Environmental Fatigue Analyses for RPV and Class 1 Piping

NUREG-2191, Revision 0 provides a recommendation for evaluating the effects of the reactor water environment on the fatigue life of a set of sample reactor coolant system components. One method to satisfy this recommendation is to assess the impact of the reactor coolant environment on a sample of critical components as described in NUREG/CR-6260. Additional component locations are evaluated if they are considered to be more limiting than those considered in NUREG/CR-6260.

Environmental fatigue calculations were performed for component locations listed in NUREG/CR-6260 for the older-vintage BWR. In order to ensure that any other locations that may not be bounded by the NUREG/CR-6260 locations were evaluated, environmental fatigue screening calculations were performed for all Class 1 RPV and piping component locations that have a reported CUF value in the governing stress reports and are exposed to the reactor coolant. These environmental fatigue calculations were performed for the limiting wetted locations for each material type in each thermal zone.

These environmental fatigue analyses will be managed by the Fatigue Monitoring (R.3.1.1) program in the same manner as the ASME Section III, Class 1 fatigue analyses. The program ensures that the Environmental Assisted Cumulative Usage Factors (CUF_{en}) are maintained below the design

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limit of 1.0. If a CUF_{en} limit is approached, corrective actions are triggered to prevent exceeding the limit.

The effects of aging on the intended functions will be adequately managed by the Fatigue Monitoring (R.3.1.1) program through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.3.6 Reactor Vessel Internals Fatigue Analyses

The PBAPS reactor vessel internals were not designed in accordance with ASME Section III. However, some of the reactor vessel internals have been subsequently evaluated for fatigue using methods from ASME Section III. These evaluations are described in following subsections:

R.4.3.6.1 Generic BWR Fatigue Analyses for Various Reactor Vessel Internal Components

The PBAPS Extended Power Uprate (EPU) license amendment submittal to the NRC in 2012 and the Measurement Uncertainty Recapture (MUR) Power Uprate license amendment submittal to the NRC in 2017 documented generic BWR fleet 40-year and 60-year CUF values for various reactor vessel internal components. This included the shroud, the core plate, the top guide, the jet pump (riser brace), the core spray line (in-vessel), the core spray sparger, and the feedwater sparger. These generic CUF values were calculated by GEH for the BWR fleet and are bounding for PBAPS. The various generic CUF values were used in the submittals to demonstrate that PBAPS reactor vessel internal components are structurally qualified for operation in the EPU and MUR operating conditions for a 60-year plant life.

The generic 40-year design CUF values were calculated by GEH in BWR fleet fatigue analyses, which assumed various normal, upset, emergency, or faulted transient severities and numbers of transient cycles. Generic 60-year CUF values were calculated by multiplying the generic 40-year design CUF values by 1.5.

These generic analyses have been identified as TLAAs and have been reevaluated for the second period of extended operation. Eighty-year CUF values have been determined for the core plate, the top guide, the jet pump (riser brace), the core spray line (in-vessel), the core spray sparger, and the feedwater sparger. All of these 80-year CUF values are projected to remain less than the acceptance criterion of 1.0 through the second period of extended operation. Section R.4.3.6.2 addresses the "Shroud" component.

These various PBAPS reactor vessel internal generic fatigue analyses have been successfully projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.3.6.2 Generic BWR Fatigue Analyses for the Core Shroud

The PBAPS Extended Power Uprate (EPU) license amendment submittal to the NRC in 2012 and the Measurement Uncertainty Recapture (MUR) Power Uprate license amendment submittal to the NRC in 2017 documented generic

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40-year and 60-year CUF values of 0.593 and 0.89 respectively for the reactor vessel core shroud. The generic 40-year CUF value was generated by GEH in a generic BWR fleet fatigue analysis. The generic 60-year CUF value was calculated by GEH by multiplying the original 40-year design CUF values by 1.5. Since this generic CUF fatigue analysis was used to demonstrate that the PBAPS core shrouds are structurally qualified for operation in the EPU and MUR operating conditions for a 60-year plant life it was identified as TLAA and reevaluated for the second period of extended operation.

The original 40-year GEH generic fatigue evaluation, which was intended to bound all BWR/4 and BWR/5 plant shrouds, used a worst-case approach to define generic the reactor vessel geometry, and thermal and mechanical stresses. The original 40-year GEH generic fatigue analysis assumed:

- 15 cycles of Cooldown - Loss of AC Power Natural Circulation Restart;
- 10 cycles of Cooldown - LPCI During Vessel Startup & Shutdown; and
- 10 cycles of Operational Basis Earthquake (OBE).

For a 60-year plant life the above assumed number of transient cycles was increased by a factor 1.5.

As of May 2017, PBAPS Units 2 and 3 have not experienced any of these transient cycles, therefore it is not credible that either PBAPS unit will experience these numbers of cycles over the remaining 80-year operating period. Therefore, the 60-year generic CUF value of 0.89 is conservative for 80 years of operation for both PBAPS units and the generic fatigue analysis remains valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.3.6.3 Core Shroud Support Fatigue Analysis Reevaluation

As discussed in the first PBAPS License Renewal Application, the core shroud support fatigue analysis was reevaluated in 1998 for the effects of increased recirculation pump starts with the recirculation loops outside of their thermal limits specifically for PBAPS. This reevaluation has been identified as a TLAA that requires evaluation for the second period of extended operation.

The SI:FatiguePro™ software includes two bounding locations representing low alloy steel and nickel alloy materials in the “Shroud Support, Baffle Plate to Vessel Junction”. These locations are bounding with respect to the cumulative fatigue usage associated with the “Core Shroud Support Fatigue Analysis Reevaluation” TLAA. The Fatigue Monitoring (R.3.1.1) program will monitor CU_{Fen} for these bounding locations and ensure corrective action is taken prior to exceeding the acceptance criterion of 1.0 through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.3.6.4 Jet Pump Diffuser/Core Shroud Support Plate Fatigue Analysis

The first PBAPS License Renewal Application projected that the 60-year CUF value for the “Jet Pump Diffuser/Core Shroud Support Plate” would not exceed a value of 0.525. This analysis has been identified as a TLAA that requires evaluation for the second period of extended operation.

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The SI:FatiguePro™ software includes a bounding location for the “Jet Pump Shroud Support, Diffuser Weld to Baffle Plate”. This location is bounding with respect to the cumulative fatigue usage associated with the “Jet Pump Diffuser/Core Shroud Support Plate” fatigue analysis TLAA. The Fatigue Monitoring (R.3.1.1) program will monitor CUF_{en} for this bounding location and ensure corrective action is taken prior to exceeding the acceptance criterion of 1.0 through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.3.6.5 Replacement Steam Dryer Stress Report and Fatigue Evaluation

To support the Extended Power Uprate (EPU) project, both PBAPS Units 2 and 3 replaced the reactor pressure vessel steam dryers. The new steam dryers were evaluated in 2014 in accordance with the requirements of ASME Section III, Division I, Subsection NG, “Core Support Structure”, 2007 Edition with 2008 Addenda. This evaluation has been identified as a TLAA that requires evaluation for the second period of extended operation.

The fatigue evaluation assumed 400 startup and shutdown transient cycles and 1 OBE event commencing in 2014. The 80-year transient projections show that the total number of startup and shutdown transient cycles will not exceed the 400 assumed in the fatigue evaluation for either unit over the entire 80-year life of each unit.

Therefore, the replacement steam dryer fatigue evaluation remains valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.3.7 High-Energy Line Break (HELB) Analyses Based on Cumulative Fatigue Usage

High-Energy Line Break (HELB) analyses for PBAPS used the CUF values from the ASME Class 1 fatigue analyses as input in determining intermediate break locations. Locations with a design CUF value less than or equal to 0.1 did not require an intermediate break to be postulated. Since the HELB analyses are based on the Class 1 piping fatigue TLAAs that provided the CUF values, the HELB analyses have been identified as TLAAs.

The current Fatigue Monitoring program uses SI:FatiguePro™ to determine the overall effect of the cumulative numbers of transient cycles that have occurred at a given time and determines the CUF values resulting from the combination of transient cycles that have occurred. SI:FatiguePro™ monitors the bounding location for all High-Energy Line Break analyses.

The Fatigue Monitoring (R.3.1.1) program will be enhanced to add a HELB CUF acceptance criterion of 0.1 for this bounding location. The Fatigue Monitoring program administrative requirements will ensure that corrective action is taken prior to the bounding location CUF value exceeding the HELB break exclusion acceptance criterion of 0.1. Therefore, these fatigue analyses will be managed by the Fatigue Monitoring (R.3.1.1) program through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

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R.4.3.8 Inservice 60-Year RPV Closure Head Weld Flaw Analyses

Volumetric examinations identified flaws on the Unit 3 closure head meridional welds CH-MA, CH-MC, and CH-MF in 2001; on the Unit 2 closure head meridional weld CH-MB in 2002; and on the Unit 2 closure head to flange weld CH-C-2 in 2010. Prior to the examinations, review of the fabrication and inspection history of these welds revealed that the indications had existed since fabrication, had previously been identified in prior ISI examinations, and most likely some of the indications would be rejected when reexamined under the new Performance Demonstration Initiative (PDI).

Three separate flaw evaluations were performed using fatigue crack growth evaluations in accordance with ASME Section XI Subsection IWB-3600 and Appendix A. The flaw evaluations assumed: 1) 100 "Bolt-Up" transient cycles, 2) 195 "Hydrostatic Test" transient cycles, and 3) 245 "Heatup-Cooldown" transient cycles and concluded that the limiting flaws were acceptable even after accounting for projected crack growth for the life of the plant including license renewal (60 total years). Therefore, these flaw evaluations have been identified as TLAAAs.

The 80-year transient cycle projections show that the total number of: "Bolt-Up", "Hydrostatic Test", and "Heatup-Cooldown" transient cycles will not exceed the number of transient cycles assumed in the flaw evaluations. Therefore, the three flaw evaluations remain valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.4 Environmental Qualification of Electric Equipment

R.4.4.1 Environmental Qualification of Electric Equipment

Thermal, radiation, and cyclical aging analyses of plant electrical and I&C components, developed to meet 10 CFR 50.49 requirements, have been identified as time-limited aging analyses (TLAAAs) for PBAPS. The NRC has established nuclear station environmental qualification (EQ) requirements in 10 CFR 50.49. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments are qualified to perform their safety function in those harsh environments after the effects of in-service aging. Harsh environments are defined as those areas of the plant that could be subject to the harsh environmental effects of a loss-of-coolant accident (LOCA), high energy line break (HELB), or post-LOCA radiation. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

The Environmental Qualification of Electric Components (R.3.1.3) program will manage the effects of aging effects for the components associated with the environmental qualification through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The program meets the requirements of 10 CFR 50.49 for the applicable electrical components important to safety. Reanalysis of an aging evaluation to extend the qualifications of components is performed on a routine basis as part of the EQ program. Important attributes

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for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, ongoing qualification, and corrective actions if acceptance criteria are not met.

If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner such that sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful.

The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The Environmental Qualification of Electric Components (R.3.1.3) program has been demonstrated to be capable of programmatically managing the qualified lives of the electrical components falling within the scope of the program for second license renewal in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.5 Concrete Containment Tendon Prestress Analysis

R.4.5.1 Concrete Containment Tendon Prestress Analysis

The PBAPS containment does not have pre-stressed tendons. As such, this topic is not a TLAA.

R.4.6 Primary Containment Fatigue Analyses

R.4.6.1 Primary Containment Structures, Penetrations, and Associated Components with Fatigue Analyses

The original design for the Primary Containment for both units was in accordance with ASME Section III, Subsection B, 1965 Edition with addenda through the Summer of 1966, which did not require an evaluation of fatigue. Subsequent to the original design, elements of the PBAPS containment were reanalyzed in response to discoveries by General Electric (GE) and others of unevaluated loads due to design basis events and Safety Relief Valve (SRV) discharge. The load definitions include assumed pressure and temperature transient cycles resulting from SRV discharge, design basis loss of coolant accident (LOCA) events, and OBE and SSE events. Components of the Primary Containment that were analyzed included the torus shell, the torus penetrations, the drywell-to-torus vents, SRV discharge piping, attached piping to the torus, and the drywell to torus vent bellows. These analyses have been identified TLAAs.

In 1997 and 1998 PBAPS Units 2 and 3 replaced the RHR and Core Spray System suction strainers in the torus. These new suction strainers and their supports were designed to ASME Section III Subsections NC, NE, and NF 1980 edition up to and including Winter 1981 Addenda and were therefore evaluated for fatigue. The suction strainer and support evaluations used as the same transient cycle inputs as the above containment analyses. Therefore, the suction strainer and support analyses have been identified TLAAs.

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The Fatigue Monitoring (R.3.1.1) program uses SI:FatiguePro™ to monitor fatigue of these components and is credited with managing these TLAAAs. The following SI:FatiguePro™ locations bound the above components for both Units 2 and 3: “Torus Penetrations (CS)/ Torus Shell” and “Torus (CS)/Torus Shell.”

The effects of aging on the intended function of these components will be adequately managed by the Fatigue Monitoring (R.3.1.1) program through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.6.2 Containment Process Line Penetration Bellows

The original PBAPS Units 2 and 3 main steam line, feedwater line, HPCI steam line, RHR supply and return line, RWCU pump suction line, core spray discharge line, and vessel head spray process line containment penetration bellows were designed to ASME Section III, 1968, Appendix IX-200, Class B Vessels and Codes Cases 1177-5 and 1330-1. The design specification for the original bellows specified 200 startup-shutdown transient cycles and a minimum of 1,500 “normal operating cycles” that would encompass Design Basis Accident (DBA) transient cycles. The Unit 3 RHR system penetration bellows were replaced in 1988 and 1989 and were designed to ASME Section III, 1980 Edition with Winter 1981 Addenda, Section NC-3649.4. The design specification for these replacement penetration bellows specified 1,500 DBA transient cycles. These analyses have been identified TLAAAs.

Transient cycle projections were performed that determined that the specified transient cycles limits will not be exceeded in 80 years. The analyses remain valid through the second period of extended operations in accordance with 10 CFR 54.21 (c)(1)(i).

R.4.7 Other Plant-Specific Time-Limited Aging Analyses

R.4.7.1 Cranes Cyclic Loading Analyses

The cranes listed below are within the scope of second license renewal and their design meets the intent of Crane Manufacturers Association of America (CMAA) Specification 70, which includes considerations for frequency of operation and expected load sizes relative to their maximum load capacity. Specification CMAA 70 allows between 20,000 and 100,000 load cycles over the service life of each crane. Therefore, 20,000 load cycles is a conservative limitation on the number of load cycles for each of these cranes. Based on these considerations, cranes are designed for a given service class classification with an expected maximum number of design load cycles over their life of the crane. The expected maximum number of design load cycles over the life of each crane provides the basis for the TLAA evaluation.

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Reactor Building Cranes

Each PBAPS unit has a reactor building crane that is within the scope of license renewal. The number of anticipated lifts for each of these cranes is estimated to be 4,032 cycles through the second period of extended operation; which is approximately 20 percent of the conservative limitation.

Emergency Diesel Generator Bridge Cranes

Each of the four Emergency Diesel Generator rooms has a bridge crane that is within the scope of license renewal. The number of anticipated lifts for each of these cranes is estimated to be 4,500 cycles through the second period of extended operation; which is approximately 25 percent of the conservative limitation.

Turbine Building Cranes

Each PBAPS unit has a turbine building crane that is within the scope of license renewal. The number of anticipated lifts for each of these cranes is estimated to be 1,400 cycles through the second period of extended operation; which is 7 percent of the conservative limitation.

Circulating Water Pump Structure Crane

PBAPS has a common circulating water pump structure crane that is within the scope of license renewal. The number of anticipated lifts for this crane is estimated to be 1,780 cycles through the second period of extended operation; which is approximately 10 percent of the conservative limitation.

These analyses remain valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

R.4.7.2 Reactor Vessel Main Steam Nozzle Clad Removal Corrosion Allowance

A 1974 analysis, developed to justify removal of the PBAPS Units 2 and 3 reactor vessel main steam nozzle cladding, used a time-dependent corrosion rate over 40 years and a corrosion allowance as acceptance criteria. As such, the evaluation was identified as a TLAA in the first License Renewal Application (LRA) that must be reevaluated for the second period of extended operation.

The evaluation concluded that a total general wall loss of 0.065 inches is an acceptable corrosion allowance for the reactor vessel main steam nozzles. Section 4.7.1 of PBAPS's first LRA documented that internal surfaces the reactor vessel main steam nozzles will only experience general wall loss of 0.030 inches over a 60-year period, which is significantly less than the corrosion allowance acceptance criterion of 0.065 inches. Based on the same corrosion rate the 80-year total general wall loss is projected to be 0.040 inches and significantly less than the corrosion allowance acceptance criterion of 0.065 inches.

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The corrosion allowance analysis has been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.7.3 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"

NUREG-0619 was issued by the NRC per Generic Letter (GL) 81-11 in February 1981 because of observed cracking on the inside surfaces of BWR feedwater nozzles at the blend radius and bore. One of the causes of the cracking was determined to be leakage between the feedwater nozzle and the thermal sleeve which resulted in rapid thermal cycling, on the order of a few cycles per second. Such cracking was observed in the feedwater nozzles at PBAPS Units 2 and 3 early in each unit's life before 1980. The cracks initiated by this rapid thermal cycling fatigue mechanism were small shallow internal cracks. However, these small cracks could potentially propagate to larger cracks by low cycle fatigue due to normal plant transients such as heatup, cooldown, and feedwater transient cycles.

In 1980 and 1981, the following three modifications recommended within NUREG-0619 were implemented on PBAPS Units 2 and 3 to reduce or eliminate the feedwater nozzle rapid thermal cycling fatigue cracking mechanism: (a) installation of improved nozzle triple thermal sleeves with dual piston ring seals, (b) removal of cladding from the nozzle bore and blend radii, and (c) improvement of the low-flow feedwater controllers. Also, the control rod drive return line (CRDRL) nozzles have been capped to eliminate cracking due to rapid thermal cycling. The CRDRL nozzles are inspected in accordance with the ASME Code, Section XI, Subsection IWB, Tables IWB-2500. The CRDRL nozzle-to-cap weld examinations are performed at a frequency specified by the BWR Stress Corrosion Cracking (R.2.1.5) program that implements commitments from NRC Generic Letter 88-01 and BWRVIP-75-A. The implementation of the modifications and improvements to plant operations at low flow conditions in the early 1980's were effective in preventing additional rapid thermal cycle fatigue cracking in the PBAPS Units 2 and 3 feedwater nozzles.

In 1983, augmented ISI inspections of the feedwater nozzles were implemented on PBAPS Units 2 and 3 based on the recommendations in NUREG-0619. The inspections depended on a fracture mechanics analyses that were used to determine inspection criteria and intervals. In 1998, PBAPS performed a new plant-specific fracture mechanics analysis in accordance with the BWROG and NRC approved guidance which was accepted by the NRC as an acceptable alternative to NUREG-0619. Since this issue was identified as a TLAA for the first license renewal, it has been reevaluated for the second period of extended operation.

The first PBAPS LRA credited "Augmented Inspections" in accordance with the Inservice Inspection (ISI) Program for the management of rapid thermal cycling fatigue cracking in feedwater nozzles. The inspections would, in effect, provide confirmation that cracking due to rapid thermal cycling fatigue or low cycle fatigue is not occurring. In addition, the first PBAPS LRA credited the Fatigue

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Management Activities Program for aging management of low cycle fatigue in the feedwater nozzles. This approach was evaluated by the NRC in NUREG-1769 and was found acceptable.

However, UFSAR Appendix C, Section C.5.3.1.1 documents that the feedwater nozzles have been evaluated for low cycle fatigue in accordance with ASME Section III as Class 1 components through the first period of extended operation. While the PBAPS plant-specific fracture mechanics analysis for rapid cycle thermal fatigue is not a TLAA and is no longer used as the basis for inspection frequency, the ASME Section III fatigue analysis of the feedwater nozzles is considered a TLAA and must be evaluated for the second period of extended operation.

Augmented Inspections in accordance with the PBAPS ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (R.2.1.1) program is credited for the management of rapid thermal cycling fatigue cracking in feedwater nozzles. These inspections provide confirmation that cracking due to rapid thermal cycling fatigue or low cycle fatigue is not occurring through the second period of extended operation. Therefore, the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (R.2.1.1) program is credited for managing rapid thermal cycling fatigue through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

In addition, the Fatigue Monitoring (R.3.1.1) program currently uses SI:FatiguePro™ to monitor low cycle fatigue of feedwater nozzles by computing the CUF_{en} values based upon the cumulative numbers of fatigue transient cycles. The SI:FatiguePro™ locations “Feedwater Nozzles (LAS)/Beyond Second Piston Ring” and “Feedwater Nozzles (CS)/Safe End Between First and Second Piston” will monitor fatigue for the feedwater nozzles to ensure that EAF usage factors will remain less than 1.0 through the second period of extended operation. Therefore, the Fatigue Monitoring (R.3.1.1) program is credited with managing the feedwater nozzle low cycle fatigue through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

R.4.7.4 Fracture Mechanics analysis of ISI-Reportable Indications for Group I Piping: As-forged Laminar Tear in a Unit 3 Main Steam Elbow Near Weld 1-B-3BC-LDO Discovered During Preservice UT

A preservice ultrasonic volumetric examination (UT) discovered an imbedded as-forged laminar indication in an elbow on a PBAPS Unit 3 main steam line. The indication did not extend into a weld. Although this portion of the main steam system piping is not subject to an ASME III Class 1 fatigue analysis, a stress and a 40-year-life fatigue analysis was performed. The analysis concluded that the primary, secondary, and primary plus secondary stresses and cumulative usage factors met ASME III requirements. As such, the evaluation was determined to be a TLAA in the first License Renewal Application which was reevaluated for the second period of extended operation.

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The original analysis calculated a 40-year worst case cumulative usage factor (CUF) of 0.036; assuming the laminar indication extends into the weld. A CUF of value 1.0 is considered the approximate threshold at which a fatigue crack may initiate and propagate. The first PBAPS LRA projected this to a maximum CUF of 0.054 for 60 years. In the SER for PBAPS's first LRA (NUREG-1769) the staff concluded that: "By reporting that the CUF is considerably below the design limit of 1.0, the staff concludes that the applicant has provided reasonable assurance that the flaw will not propagate during the 40-year life of the plant and the period of extended operation." Therefore, over an 80-year period the maximum CUF is projected to be 0.072 which is less than the ASME Section III cumulative fatigue acceptance criterion of 1.0.

The fatigue analysis has been projected through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

R.4.7.5 Unit 3 Core Spray Replacement Piping Fatigue and Leakage Assessment

In 2013, portions of the Unit 3 Core Spray System piping segment located in the reactor vessel, from the N5A and N5B nozzle thermal sleeves to the shroud wall were replaced. Some of the mechanical replacement hardware introduced a number of very small openings in the new piping segment and its attachment to the shroud wall, thus providing leakage paths. Therefore, a leakage assessment was performed to demonstrate that the new piping would perform its intended function through the remaining life of the unit. The leakage assessment assumed corrosion rates over 40 years. The 40-year service life of the Core Spray System piping segment ends in 2053, and the second period of extended operation will end in 2054 for Unit 3.

In addition, a fatigue analysis of the new piping system and associated bolting concluded a maximum design CUF value of 0.0565. The fatigue analysis assumed the following transients starting in 2013: 161 cycles of Startup and Shutdown; 197 cycles of Scram: Turbine Trip and All Other Scrams; 80 cycles of Loss of Feedwater; 40 cycles of Scram – Single Relief or Safety Valve Blowdown; 50 Cycles of Injections; and one OBE with 50 cycles. Since the leakage assessment and fatigue analysis assumed a 40-year service life they have been identified as TLAAAs.

The leakage assessment was reevaluated for an additional five years for a total of 45 years, until 2058. The reevaluation determined that the conclusions of the original leakage assessment are valid for an additional five years. The associated fatigue analysis was also reevaluated for an additional five years of service life for a total of 45 years, until 2058. The reevaluation concluded that the originally assumed number of transient cycles, shown above; remain valid for 45 years and the design CUF value of 0.0565 remains unchanged.

Therefore, both the leakage assessment and the fatigue analysis remain valid through the second period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

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R.5.0 SECOND LICENSE RENEWAL COMMITMENT LIST