Exelon Generation.

10CFR 50.59, 10CFR 72.48, NEI 99-04 (SECY 00-0045)

January 27, 2017

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station (PBAPS), Units 1, 2 and 3 and
PBAPS Independent Spent Fuel Storage Installation (ISFSI)
Facility Operating License No. DPR-12
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-171, 50-277, 50-278, and 72-29 (ISFSI)

Subject: Biennial 10CFR 50.59 and 10CFR 72.48 Reports for the Period 1/1/2015 through 12/31/2016 and Annual Commitment Revision Report for the Period 1/1/16 through 12/31/16

Attached are the 2015-2016 Biennial 10CFR 50.59 and 10CFR 72.48 Reports and the 2016 Annual Commitment Revision Report as required by 10CFR 50.59(d)(2), 10CFR 72.48, and SECY-00-0045 (NEI 99-04).

There are no new regulatory commitments contained in this transmittal.

If you have any questions or require additional information, please contact D. J. Foss at 717-456-4311.

Sincerely,

Pat D. Navin

Patrick D. Navin
Plant Manager
Peach Bottom Atomic Power Station

cc: Senior Resident Inspector, USNRC, PBAPS
Commonwealth of Pennsylvania
Document Control Desk, USNRC, Washington DC

CCN: 17-02

Attachment

Attachment

Exelon Nuclear
Peach Bottom Atomic Power Station

Docket Nos. 50-171
50-277
50-278
72-29

2015-2016 Biennial 10CFR 50.59 and 10CFR 72.48 Reports and the 2016 Commitment Revision Report

These reports are issued pursuant to reporting requirements for Peach Bottom Atomic Power Station Units 1, 2 and 3. These reports address tests and changes to the facility and procedures as they are described in the Peach Bottom Final Safety Analysis Report and Independent Fuel Storage Safety Analysis Report for the TN-68 Spent Fuel Cask. These reports consist of those tests and changes that were implemented between January 1, 2015 and December 31, 2016. Also, this report identifies commitments that were revised during 2016 and require reporting in accordance with the guidelines of NEI 99-04, Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff endorsed by SECY-00-0045.

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Exelon Nuclear
Peach Bottom Atomic Power Station
Units 1, 2 and 3
Docket Nos. 50-171, 50-277, and 50-278

BIENNIAL 10CFR 50.59 REPORT
JANUARY 1, 2015 THROUGH DECEMBER 31, 2016
EVALUATION SUMMARIES

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Title: **Condensate Storage Tank (CST) Standpipe and Cross-Connect Pipe Addition (ECR 11-00378)**

Units Affected: **3**

Year Implemented: **2015**

Tracking No.: **PB-2013-002-E, Rev. 1**

Brief Description:

This activity modified the CST by adding a standpipe in the tank. The standpipe will prevent draining of CST to the condenser hotwell in the event of spurious opening of the hotwell makeup valves. Additionally, a cross connect pipe was added between the Unit 3 High Pressure Coolant Injection (HPCI) / Reactor Core Isolation Cooling (RCIC) suction piping from the CST and hotwell makeup/reject piping. Under Extended Power Uprate (EPU) conditions, the CST inventory dedicated to HPCI and RCIC suction is credited for Station Blackout (SBO), Anticipated Transient without Scram (ATWS) and Appendix R events and therefore, this modification preserves the availability of these systems. This activity does not impact plant operations at nominal CST levels; however, a new action of opening existing manually operated isolation valves is required to allow the CSTs to continue to perform their design function of providing a backup water supply to the CRD pumps when CST inventory is below the height of the installed standpipe.

Summary of Evaluation:

The installation of the standpipe in the CST and the addition of the cross connect pipe do not adversely impact design bases or safety analyses as described in the UFSAR. This activity does not adversely impact plant operations, design bases or safety analyses as described in the UFSAR.

The 50.59 Evaluation determined that this change did not increase the frequency or consequences of a previously evaluated accident or create the possibility of a new accident since no accident initiators are involved. It does not increase the likelihood of occurrence of a previously evaluated malfunction of an SSC important to safety because the affected equipment does not interfere with any previously evaluated. It does not increase the consequences of a previously evaluated malfunction of equipment important to safety because there are no consequences associated with the activity. It does not create the possibility for a malfunction of

an SSC important to safety with a different result than any previously evaluated in the UFSAR because no new failure modes are introduced. It does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered because no system parameters will change as a result of this activity.

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Title: **Reactor Feedpump Turbine (RFPT) Replacement – Electrical / Instrumentation (ECR 13-00266)**

Units Affected: **3**

Year Implemented: **2015**

Tracking No.: **PB-2014-002-E, Rev. 0**

Brief Description:

In conjunction with the Reactor Feed Pump Turbine (RFPT) replacement, the mechanical overspeed device and trip mechanism was replaced with an electrical overspeed device and trip module. This change from functionally diverse to functionally equivalent overspeed protection altered the means of performing this function and affects a design function of the Turbine Driven Feedwater Pump Control as described in UFSAR Section 7.10.4.

As a result of this activity, the primary overspeed function is performed by a new protection device that is functionally equivalent to the previous device. While both devices are microprocessor based, they are electrically diverse and not subject to a common mode failure. Using guidance provided in NUREG/CR-6303 (Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems), the new overspeed protection device is design diverse (different architecture), equipment diverse (same manufacturer of fundamentally different designs), signal diverse (same parameter sensed by different sensors) and software diverse (different program architecture) as compared to the previous system. As such, the same defense-in-depth will be provided by the electrically diverse redundant overspeed devices as with the existing functionally diverse overspeed devices.

Summary of Evaluation:

The facility change will not alter the manner in which the RFPT, RFPT Speed Control, Feedwater Control, Lube Oil or 125 VDC systems are controlled or operated. The same monitoring and protective functions will be performed by the modified system as currently performed by the existing RFPT instrumentation and controls. This facility change does not affect any Nuclear Safety Related components.

With the same defense-in-depth, the facility change does not increase the likelihood of a malfunction of equipment important to safety or the frequency of accidents evaluated in the UFSAR. Although the new digital equipment has different modes of failure, the effect of these failures is the same and does not create the possibility of a different accident or the malfunction of equipment important to safety with a different result.

No new system interfaces are created and no physical changes are made to a steam path or barrier that could alter or affect the consequences of an accident. The radiological consequences

of the malfunctions and accidents currently evaluated are not affected and are bounding for this facility change.

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Title: **Digital Electro-Hydraulic Control (DEHC) Modification**

Units Affected: **2**

Year Implemented: **2016**

Tracking No.: **PB-2015-001-E, Rev. 0**

Brief Description:

This configuration change implements an upgrade to the Pressure Regulator and Turbine-Generator Control System described in UFSAR Section 7.11 for Unit 2. This upgrade will replace the analog Turbine Control System with a digital Turbine Control and Protection System (referred to as the DEHC System). The DEHC system utilizes a distributed control system that includes a Turbine Control System (TCS) and an Emergency Trip System (ETS), each consisting of redundant controllers, power supplies, I/O and testable dump assemblies. The DEHC System TCS performs the reactor pressure control, turbine speed and load control, and system test functions and provides backup overspeed protection. The DEHC System ETS performs the primary turbine over-speed protection and all other turbine protection related functions.

Summary of Evaluation:

The turbine-generator control functions for the DEHC system (speed control, load control, flow control, valve control, or load limit, and turbine protection) are the same or bound the existing EHC system. Valve positions will be measured by the existing Linear Variable Differential Transformers (LVDT). The DEHC system function of controlling reactor pressure will be functionally the same as the existing EHC system. The operational requirements associated with reactor pressure, turbine speed, trip setpoints, load limits, generator runback, maximum combined flow limit, valve stroke times and Bypass Valve response times remain the same for the DEHC System as the previous EHC System. Although the implementation of the functions are performed in a different manner (electrical versus mechanical and use of redundant devices), the design bases described in the UFSAR are unchanged. There is no impact on how the turbine operating parameters are controlled even though the operator interface with the turbine controls is changed (i.e., HMI software controls vs. hardware controls). The failure modes of the DEHC System, including software failure modes, are equivalent to or bounded by the failure effects of the existing EHC System. The effect of the proposed change on the design functions is that the logic for these functions will now be accomplished within the DEHC system software as opposed to the relays and transistor/amplifier logic of the EHC system. The reliability of the DEHC system is improved through the use of a redundant design for critical functions.

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Title: **Replacement of Recirculation Motor-Generator Drives with an Adjustable Speed Drive (ASD) System (ECRs 13-00338 and 15-00378)**

Units Affected: **2 / 3**

Year Implemented: **Unit 3 – 2015, Unit 2 – 2016**

Tracking Nos.: **PB-2015-002-E, Rev. 0 (Unit 3) and PB-2016-003-E, Rev. 0 (Unit 2)**

Brief Description:

This activity involved the installation of Adjustable Speed Drives (ASD) to replace the existing Reactor Recirculation Motor-Generator (M-G) sets. The main components of the M-G sets replaced are the drive motors, oil filled fluid drive coupling with scoop tube and positioners, generators, oil filled heat exchangers and pumps. The ASD main components are a multiple winding step-down transformer, AC-DC variable AC power converters, water cooling system with heat exchangers and dual controlled and protective microprocessor system with independent back-up protection relays. The ASD is a solid state static three phase inverter which converts the 60 Hz input frequency to a variable output frequency to control the Reactor Recirculation motor speed as the existing M-G sets. The previous design basis (including seismic requirements) is not changed, just the method of implementing Recirculation Pump speed control.

Summary of Evaluation:

The Reactor Recirculation System is a non-safety related system that is important for power production. The basic function of the reactor recirculation system is to modulate reactor power level during normal power production operations. Start-up of the ASD is slightly different than the start-up of the M-G set due to the changes in method of control and different types of startup permissives present in the control circuits. The ability to initiate manual reactor recirculation runbacks has been added to aid Operations in system control when plant and procedural conditions require it.

The ASD uses solid state power conversion and highly reliable state-of-the art digital control systems to produce the frequency and voltage applied to the pump motor. Reactor Recirculation pump speed will be more precise, controllable, stable and reliable due to the fully redundant microprocessor based electronic control systems verses the existing obsolete and unreliable single failure vulnerable electro-mechanical control system. Replacing the M-G set and associated fluid coupling eliminated the variability, spurious fluctuations and instability experienced with the present analog control and fluid coupling technology.

A 50.59 Evaluation was required because of the need to consider the changes in the method of controlling the Reactor Recirculation pump with the use of digital interfaces with the digital ASD control and ultimately the control and monitoring of core flow. The use of an ASD using electronic frequency conversion changes the pump response time on ASD feeder breaker trip and introduces the potential for higher frequency and faster acceleration than possible with the M-G Set. However, for the transients and accidents analyzed in the Evaluation, it is the ATWS/RPT breaker that trips the recirculation pump motor and not the input breaker trip. Therefore, the faster coast down time with input breaker trip is not a factor in this analysis. Also, the internal ASD limits are backed-up by independent overfrequency trip relays to protect the motor from inadvertent overspeeding. Thus, these conditions have been analyzed and determined to be acceptable. No increases in transients, accidents, malfunctions, consequences or increased probabilities of likelihood of an accident or malfunction have been identified in the 50.59 review process.

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Title: **Interim Configuration of the 'B' Residual Heat Removal (RHR) Loop for Installation of RHR Cross-tie Modification (ECR 11-000496)**

Units Affected: **3**

Year Implemented: **Unit 3 - 2015**

Tracking No.: **PB-2015-003-E, Rev. 0**

Brief Description:

To support the implementation of Extended Power Uprate (EPU), the purpose of the RHR Cross-Tie modification and associated operational changes is to provide for increased containment cooling capability following a DBA event with Loss of Offsite Power (LOOP) and a single failure of an Emergency Diesel Generator (EDG), 4kV bus, or 125Vdc safety-related battery (results in loss of associated 4kV bus or EDG). In addition, this modification will limit the maximum Low Pressure Core Injection (LPCI) flow rate to provide for RHR Pump NPSH margin. The interim U3 RHR System configuration installed by ECR 11-00496 adds piping connecting the B and D trains of the B RHR loop. The installed piping includes two manual isolation valves (one at the connection to each train), a motor-operated cross-tie shut-off valve, and manual vent and drain valves. In the interim configuration, the manual isolation valves will be locked in the open position to avoid thermal pressurization of the new piping. The manual vent and drain valves will be closed. Finally, the motor-operated cross-tie shut-off valve will be locked in the closed position and there will be no electrical power connected to it. This locked-closed cross-tie shut-off valve provides the separation of the B and D trains of the B RHR loop.

Summary of Evaluation:

The evaluation showed that the adverse effects associated with the change to the UFSAR-described design function of the RHR System in the SPC mode of operation introduced by the U3 RHR System interim configuration does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated, does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety, does not create a possibility for an accident of a different type, does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety or accident, previously evaluated in the UFSAR. The proposed activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. The modified system for the Interim and Contingent configurations is functionally equivalent to the current configuration.

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Title: **Technical Requirements Manual (TRM) Test Requirement (TR) 3.5.4 Test Interval Change for Containment Spray – Containment High Pressure (Function 5)**

Units Affected: **2 / 3**

Year Implemented: **2015**

Tracking No.: **PB-2015-004-E, Rev. 0**

Brief Description:

The change to TRM 3.5 involved a testing interval change for surveillance tests that verify operability of the high drywell signal inputs into the Low Pressure Coolant Injection (LPCI) system, an operating mode of Residual Heat Removal (RHR), and Containment Spray. The change to the frequency of testing of these functions does not affect plant operations, the system design basis, or any safety analyses described in the UFSAR. The valves that allow the diversion of cooling water for Containment Spray are automatically closed upon receipt of an LPCI initiation signal as designed to provide cooling water to the reactor vessel following the design basis loss of cooling accident (LOCA). The manual controls for the drywell valves to initiate containment spray are interlocked so that opening the valves by manual action is not possible unless both primary containment (drywell) pressure is high, which indicates the need for containment cooling, and reactor vessel water level inside the core shroud is above the level equivalent to two-thirds the core height. The Containment Spray – Containment High Pressure function is a permissive to ensure adequate cooling is maintained for the reactor vessel during a design basis LOCA and prevents diversion of cooling water to Containment Spray. This activity only affects the frequency of testing the Containment High Pressure function for Containment Spray and does not affect the scope nor the frequency of testing the function as discussed in the UFSAR other than changing the TRM TR 3.5.4 frequency of 92 days to 12 months.

Summary of Evaluation:

An Engineering evaluation documented in the 50.59 evaluation supported that the impact from the test interval change remains below thresholds for concern and there is a negligible increase in failure of any SSCs. The Containment Spray function, as described in the UFSAR, is to provide additional redundancy for Containment Cooling for post-accident conditions. The valves that allow the diversion of cooling water for Containment Spray are automatically closed upon receipt of an LPCI initiation signal as designed to provide cooling water to the reactor vessel following the design basis loss of cooling accident (LOCA). The manual controls for the drywell valves to initiate containment spray are interlocked so that opening the valves by manual action is not possible unless both primary containment (drywell) pressure is high, which indicates the need for containment cooling, and reactor vessel water level inside the core shroud is above the level equivalent to two-thirds the core height. The activity does not change design bases requirements as described in the UFSAR. The test interval change to the TR does not introduce a change to the system that would prevent it from performing its intended function. The proposed activity does not degrade or inhibit the capability of the plant to cope with the conditions included in the design basis. This activity does not introduce any new failure modes.

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Title: **Reactor Feed Pump Turbine (RFPT) Trip Logic Modification (ECRs 15-00260 and 16-00027)**

Units Affected: **2 / 3**

Year Implemented: **Unit 3 – 2015, Unit 2 – 2016**

Tracking Nos.: **PB-2015-006-E, Rev. 0 (Unit 3) and PB-2016-002-E, Rev. 0 (Unit 2)**

Brief Description:

The RFPT control logic was modified for the condenser low vacuum trip. The removal of the Low Condenser Vacuum trip logic directly affects plant operations since a manual trip on the main control room alarm will now be required. Procedures were revised to implement this activity. The RFPT Vacuum Trip Bypass is no longer required for startup since the automatic trip is removed; the pushbutton, indicating light, relay and associated wiring was removed/abandoned. Previously, the logic provided an automatic trip function for the RFPT on decreasing vacuum at 20"HGV.

Summary of Evaluation:

The removal of the automatic RFPT trip on Low Condenser Vacuum adversely affects UFSAR described design functions described in Section 7.10.4. The proposed activity altered the turbine protection method by removal of the automatic turbine trip on Low Condenser Vacuum from the RFPT trip logic. The requirement for a trip remains and will be performed manually by the Operator if the alarm from the exhaust vacuum pressure switch is initiated. Also, this change eliminates the function for the RFPT Vacuum Trip Bypass Switch and the switch/associated components/wiring are removed by this ECR. There is no change to any procedure that adversely affects how UFSAR described design functions are performed, nor is there any change to a UFSAR described evaluation methodology or is an alternative methodology used for establishing design bases or in the safety analyses. The proposed activity does not involve any tests or experiments not described in UFSAR nor does it utilize/control an SSC in a manner outside the reference bounds of the SSC. The 50.59 Evaluation determined that this change did not increase the frequency or consequences of a previously evaluated accident or create the possibility of a new accident since no accident initiators are involved. It does not increase the likelihood of occurrence of a previously evaluated malfunction of an SSC important to safety because the affected equipment does not interfere with any previously evaluated. It does not increase the consequences of a previously evaluated malfunction of equipment important to safety because there are no consequences associated with the activity. It does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR because no new failure modes are introduced. It does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered because no system parameters will change as a result of this activity.

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Title:	**Local Leak Rate Testing Scope Reduction for Residual Heat Removal and Core Spray (ECR 15-00314)**
Units Affected:	**2 / 3**
Year Implemented:	**2015**
Tracking No.:	**PB-2015-007-E, Rev. 0**

Brief Description:

This activity invokes an exemption from the performance of 10CFR50 Appendix J, Type C testing for certain Primary Containment Isolation Valves (PCIVs) associated with the Residual Heat Removal (RHR) System Shutdown Cooling and Low Pressure Coolant Injection (LPCI) isolation valves, and Core Spray (CS) System isolation valves. Removal of Local Leak Rate Testing (LLRT) for valves which meet the exemption criteria will reduce personnel radiation exposure and will decrease work scope for refueling outages.

Technical Specification 5.5.12, Primary Containment Leakage Rate Testing Program states in part that "A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, Performance-Based Containment Leak-Test Program, dated September 1995, as modified by the following exceptions to NEI 94-01, Rev. 2A and 3A, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J." In NEI 94-01 Revision 2A and 3A, Section 6.0, General Requirements, components that may be exempted from the LLRT requirements of 10CFR50 Appendix J include primary containment boundaries that do not constitute potential primary containment atmospheric pathways during and following a Design Basis Accident (DBA). An Engineering evaluation established that both the RHR and CS systems are closed systems outside containment, as confirmed by UFSAR section 4.8.2.3.

Summary of Evaluation:

This activity eliminated the currently performed 10CFR50, Appendix J, Type C leakage testing for the PCIVs listed in the description of activity above. This elimination is in compliance with Regulatory, Industry Standards and Exelon Procedure for exclusion from LLRT. The subject PCIVs which are also designated as Pressure Isolation Valves (PIVs) will continue to be leak tested as PIVs in accordance with In-Service Testing (IST) Program required by 10CFR50.55a. A UFSAR change was required to update UFSAR Table 5.2.2, Containment Penetrations Compliance with 10CFR50, Appendix J to add discussion that the isolation provisions consist of a water-filled closed system outside containment and therefore, 10CFR50, Appendix J testing is not required since a potential primary containment atmospheric pathway does not exist. The 50.59 Evaluation determined that this change did not increase the frequency or consequences of a previously evaluated accident or create the possibility of a new accident since no accident initiators are involved. It does not increase the likelihood of occurrence of a previously evaluated malfunction of an SSC important to safety because the affected equipment does not interfere with any previously evaluated. It does not increase the consequences of a previously evaluated malfunction of equipment important to safety because there are no consequences associated with the activity. It does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR because no new failure modes are introduced. It does not result in a design basis limit for a fission product barrier as described in

the UFSAR being exceeded or altered because no system parameters will change as a result of this activity.

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Title: **Technical Requirements Manual (TRM) Test Requirement (TR) 3.5.5 Test Interval Change for RHR-LPCI Injection Valve Closure Permissive – Reactor Low Pressure (Function 3)**

Units Affected: **2 / 3**

Year Implemented: **Unit 3 – 2015, Unit 2 – 2016**

Tracking Nos.: **PB-2015-008-E, Rev. 0 (Unit 3) and Rev. 1 (Unit 2)**

Brief Description:

This activity involves changing the testing interval of surveillance tests (ST) that verify operability of the Low Pressure Coolant Injection (LPCI) Injection Valve Closure Permissive – Reactor Low Pressure function, which permits closure of the RHR LPCI injection valves in conjunction with a primary containment isolation signal. The change to the channel calibration test interval is from 92 days (Technical Requirement (TR) 3.5.5) to 24 months (TR 3.5.6). The pressure switches were replaced with transmitters and trip units. Performing the STs in this new configuration could result in a unit scram. Therefore, they will be performed during unit outages (24 month frequency) to eliminate the potential for causing an online scram during performance of the STs.

Summary of Evaluation:

The change to the frequency of testing does not affect plant operations, the system design basis, or any safety analyses described in the UFSAR. The change to the channel calibration test interval for Low Pressure Coolant Injection (LPCI) Injection Valve Closure Permissive – Reactor Low Pressure from 92 days (TR 3.5.5) to 24 months (TR 3.5.6) has been evaluated and does not result in a more than minimal increase in the likelihood of occurrence or consequences of a malfunction of the subject instrumentation or any SSC important to safety since the subject plant equipment has been evaluated for the new testing interval and no change to the RHR system design is involved. The proposed activity does not result in a more than minimal increase in the frequency or consequences of an accident previously evaluated in the UFSAR. Since there is no change to any plant function as a result of this activity, no possibility of a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR is created. Similarly, no possibility of an accident of a different type than any previously evaluated in the UFSAR is created. This activity has no effect of any kind on any design basis limit for a fission product barrier.

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Title: **Compensatory Measures for Operability for 2K Automatic Depressurization System (ADS) Safety Relief Valve (SRV) (ECR 16-00114)**

Units Affected: **2**

Year Implemented: **2016**

Tracking Nos.: **PB-2016-001-E, Rev. 0**

Brief Description:

This activity involves the temporary implementation of Compensatory Measures for an Operability Evaluation as documented in ECR 16-00114. The Operability Evaluation has determined that the ADS logic function of the 2K Safety Relief Valve (SRV) is degraded, but OPERABLE to meet the requirements of TS 3.3.5.1. This OPERABLE determination was primarily based on the fact that the DC system at PBAPS is an ungrounded system that is designed to tolerate a system ground. However, the Operability Evaluation identified that certain actions would be prudent as Compensatory Measures to prevent inadvertent SRV actuations if a second ground were to occur. The temporary compensatory actions have no impact on the automatic overpressure relief function of the 2K SRV. This activity is being performed to prevent inadvertent actuations of the 2K SRV similar to what occurred on 3/14/16 (IR 2640316). The SRV actuation occurred during the racking in of the 2A Control Rod Drive (CRD) 4 kV breaker during clearance restoration. This evolution resulted in a hard ground to the DC system, which coupled with a low resistance condition in the 2K SRV control circuit caused the 2K SRV to actuate twice. As a result of the inadvertent SRV actuations, the ADS function of the SRV was declared inoperable and the control fuses removed to prevent further undesirable actuations. Although the hard ground was repaired, the SRV was maintained inoperable pending additional actions to ensure that additional inadvertent actuations would not occur.

The implementation of this temporary activity resulted in more robust 2K SRV ADS reliability and management of grounds in the Division 1 (2A/2C) DC subsystem that could adversely impact the 2K SRV, until permanent repairs were completed.

Summary of Evaluation:

The 10CFR 50.59 applicability review determined that further screening was required since this is a temporary change to a UFSAR described design function. The 10CFR 50.59 screening determined that an adverse change existed since additional contacts were being added in the 2K SRV control logic. These contacts are required to close to perform the ADS and remote-manual function of the 2K SRV. No changes to Technical Specifications or the Operating License were involved with the activity. The 10CFR 50.59 Evaluation determined that there was not more than a minimal increase in the probability of malfunction of the associated equipment. There were no increases in the probability of an accident. No new accidents were created. No different results were created by an assumed malfunction. There were no effects on the dose consequences, methods of evaluation or design basis limits of fission product barriers. Based on the above, there is no NRC approval required to implement this activity.

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Title: **Technical Requirements Manual Section (TRMS) 3.11 Change for Applicability of Residual Heat Removal (RHR) Cooler / Fan Units (ECR 16-00241)**

Units Affected: **2 / 3**

Year Implemented: **2016**

Tracking No.: **PB-2016-006-E**

Brief Description:

This activity involves a revision to Technical Requirements Manual Specification (TRMS) Section 3.11, Engineered Safeguards (ES) Compartment Cooling and Ventilation (and its associated TRMS Bases sections). The change is limited to the Residual Heat Removal (RHR) cooler / fan unit portion of the TRMS 3.11. Currently, the TRMS applicability is 'when the associated pumps are required to be OPERABLE'. The applicability is being revised to limit TRMS 3.11 OPERABILITY requirements for the RHR cooler / fan units to not include the RHR cooler/fan units if in Modes 4 or 5 with RHR suction source water temperature < 110^{O}F. RHR cooler / fan units will not be required to be OPEARBLE for these conditions for the purposes of supporting Technical specification (TS) 3.5.2., TS 3.4.8, TS 3.9.7 and TS 3.9.8.

Summary of Evaluation:

The implementation of this activity allowed for continued OPERABILITY of RHR during shutdown conditions during partial outages of the Emergency Service Water (ESW) that affect the Emergency Core Cooling Systems. An Engineering Calculation has been completed that has determined that not having ESW / normal Service Water (SW) available during design basis conditions postulated during Modes 4 and 5 will not result in the inoperability of RHR subsystems required by TS 3.5.2, TS 3.4.8, TS 3.9.7 and TS 3.9.8 as long as the RHR suction source water temperature < 110^{O}F. This evaluation concluded that the loss of cooling to the cooler / fan units in the RHR rooms would not result in the inoperability of the RHR subsystems. The applicable design conditions assumed a concurrent, seismic event, Loss-of-Offsite-Power (LOOP) condition RHR suction source water temperature < 110^{O}F and the plant operating in Mode 4 or 5.

The 10CFR 50.59 applicability review determined that further screening was required since this is a change to a UFSAR described design function of the Engineered Safeguards cooler / fan units. The 10CFR 50.59 screening determined that an adverse change existed since the RHR subsystems would be challenged by higher heat loads than would otherwise have existed without this change activity. However, it was determined that RHR TS 3.5.2, TS 3.4.8, TS 3.9.7 and TS 3.9.8 subsystems would still be OPERABLE. No changes to Technical Specifications or the Operating License were involved with the activity. The 10CFR 50.59 Evaluation determined that there was not more than a minimal increase in the probability of malfunction of the associated equipment. There were no increases in the probability of an accident. No new accidents were created. No different results were created by an assumed malfunction. There were no effects on the dose consequences, methods of evaluation or design basis limits of fission product barriers.

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Title: **Changes to Analytical Limits in the Neutron Monitoring System (ECR 16-00248)**

Units Affected: **2 / 3**

Year Implemented: **2016**

Tracking No.: **PB-2016-007-E, Rev. 0**

Brief Description:

This activity involves changing the Analytical Limits (AL) for certain setpoint functions associated with the Neutron Monitoring System. The Neutron Flux-High scram AL is changed from 122.0% Rated Thermal Power (RTP) to 125.0% RTP. The Neutron Flux-High (Setdown) scram AL is changed from 17.3% RTP to 21.0% RTP. For the Rod Block Monitors (RBM), the ALs for the High Power Trip Setpoint (HTSP), Intermediate Power Trip Setpoint (ITSP) and Low Power Trip Setpoint (LPSP) are increased by nominally 2% RTP; the exact magnitudes of the changes are dependent on the cycle-specific Maximum Critical Power Ratio (MCPR).

This activity does not impact any Technical Specifications Allowable Values (AV) or involve any field work. The affected functions are listed in Technical Specifications Table 3.3.1.1-1 (functions 2a and 2c) for the scrams and Table 3.3.2.1.-1 (functions 1 a, 1 b, le) for the rod blocks. Testing performed at Peach Bottom Units 2 and 3 after the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) installations identified higher than anticipated neutron noise signals. These noise levels were not a direct result of the MELLLA+ implementation. Neutron noise is a parameter that factors into the Neutron Monitoring System instrument uncertainty calculation. A neutron noise level of 1.25% Rated Thermal Power (RTP) was used in the calculation, but testing after the MELLLA+ installation identified worst-case noise levels slightly below 3.1 % RTP. This was considered a non-conforming condition since higher than analyzed noise levels were measured in the plant. Analyses were performed which determined that increasing the ALs for the impacted setpoints to compensate for the larger neutron noise factor is an acceptable approach.

Summary of Evaluation:

By increasing the ALs to compensate for the margin lost due to the increased neutron noise factor, the existing Technical Specification AVs and field settings can be maintained as-is. All applicable safety analyses remain valid when the new ALs are used as inputs. This activity has no effect on plant operations as no field changes are involved. The ALs being changed are theoretical design values used in various analyses. The ALs are being raised because neutron noise during MELLLA+ testing was higher than that modeled in the setpoint calculation of record. Analyses related to the Neutron Flux-High scrams (Technical Specifications Table 3.3.1.1-1, functions 2a and 2c) that were evaluated for impact included the over-pressurization protection safety analysis, the pressure regulator failure-downscale (PRFDS) with the backup pressure regulator out of service (PROOS) event, the Control Rod Drop Accident (CRDA) and all other Anticipated Operational Occurrences. For the Rod Block Monitor (Technical Specifications Table 3.3.2.1-1), the licensing thermal-mechanical compliance conclusions for the Rod Withdrawal Error (RWE) event have been confirmed to remain applicable. In summary, no conclusion of any safety analysis is impacted by the AL increases.

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Title: **Technical Requirements Manual (TRM) Test Requirement (TR) 3.1.4 Test Interval Change for Reactor Vessel Water Level Low-Low**

Units Affected: **2 / 3**

Year Implemented: **2016**

Tracking Nos.: **PB-2016-008-E, Rev. 0**

Brief Description:

This activity involved changing the frequency of performance of TR 3.1.4 for Reactor Vessel Water Level - Low, Low under Technical Requirements Manual (TRM) 3.1, Alternate Rod Insertion (ARI) instrumentation. TR 3.1.4 was revised to specify a calibration check every outage (24 months). This activity will reduce the frequency to every other outage (48 months) for Reactor Vessel Water Level Low-Low. This change is based on a history of reliable operation of the associated instruments. Extending the frequency will reduce dose exposure to workers and support the divisional strategy initiative by dividing workload across multiple outages.

Summary of Evaluation:

The change to the frequency of testing does not affect plant operations, the system design basis, or any safety analyses described in the UFSAR. The change to the frequency has been evaluated and does not result in a more than minimal increase in the likelihood of occurrence or consequences of a malfunction of the subject instrumentation or any SSC important to safety since the subject plant equipment has been evaluated for the new testing interval and no change to the RHR system design is involved. The proposed activity does not result in a more than minimal increase in the frequency or consequences of an accident previously evaluated in the UFSAR. Since there is no change to any plant function as a result of this activity, no possibility of a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR is created. Similarly, no possibility of an accident of a different type than any previously evaluated in the UFSAR is created. This activity has no effect of any kind on any design basis limit for a fission product barrier.

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Title: **Main Steam Isolation Valve (MSIV) 80D Poppet Skirt Modification (ECR 16-00346)**

Units Affected: **2**

Year Implemented: **2016**

Tracking Nos.: **PB-2016-010-E, Rev. 0**

Brief Description:

A modification was performed to install a skirt on the back of the poppet of the Peach Bottom Unit 2 'D' Inboard Main Steam Isolation Valve (MSIV). This skirt is intended to hold the poppet firmly against the cover to eliminate poppet vibration (IR 2733506). Excessive wear and increased leakage has been observed on this valve, caused by vibration of the valve poppet against the

valve body. The skirt will contact the valve bonnet when the valve is in the full open position, where the previous design would backseat against the stem. This will stabilize the poppet against the valve cover and prevent excessive movement of the poppet. The skirt provides a bearing area of much larger diameter than that of the stem backseat. Consequently, the skirt is capable of providing much greater stabilizing resistance to the rocking movement of the open poppet. This greater stabilization is intended to prevent excessive movement of the open poppet and associated damage to it. This modification will improve the sealing of the MSIV when it is closed. The new MSIV configuration is similar in function to a modification installed by the MSIV manufacturer, and used successfully in a number of BWRs, to prevent similar poppet damage.

Summary of Evaluation:

This activity will limit operational damage to the MSIV, in that it will stabilize the poppet and minimize wear against the valve body. Consequently, in order to stabilize the poppet, the UFSAR-described function of the valve backseat to prevent leakage through the stem packing (UFSAR 4.6.3) is voided by the proposed activity. The valve packing will be relied upon to prevent operational leakage to the drywell. There is no impact on the valve's ability to close, or the timing in which it will close. Therefore, safety analyses are unaffected by this change.

The proposed activity does not result in an increase in the frequency of occurrence or consequences of an accident or the likelihood or consequences of a malfunction. The activity does not create the possibility for an accident of a different type, or for a malfunction of an SSC with a different result. No DBLFPB's are altered or exceeded by this activity. The proposed activity does not affect the conformance with the Technical Specifications nor the conditions of the Facility Operating License. Consequently, the proposed Activity does not require a change to the Technical Specifications or Facility Operating License.

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Title: **Methods Change for Environmental Fatigue Calculations**

Units Affected: **2 / 3**

Year Implemented: **2016**

Tracking No.: **PB-2016-012-E, Rev. 0**

Brief Description:

The activity changed the methodology used to evaluate the impact of reactor water environment on the fatigue life of reactor pressure vessel components. PBAPS was previously committed to performing evaluations in accordance with NUREG/CR-6583 and NUREG/CR-5704, which was replaced by NUREG/CR-6909 Rev. 0. UFSAR section Q.5.2.4 will be revised to incorporate NUREG/CR-6909 for environmental fatigue evaluations. This methodology is already used in station evaluations and demonstrates acceptable fatigue life of reactor vessel components. This methodology is endorsed by the NRC for this application. NUREG/CR-6909 Rev. 0 draws on a larger database than was used for previous guidance, provides guidance for nickel alloy materials, and addresses non-conservatisms in earlier methodology for stainless steels. Therefore, PBAPS adopted this revised methodology for environmental fatigue evaluations.

Summary of Evaluation:

Due to a change in UFSAR-described methodology, a 50.59 Evaluation was required. This evaluation shows that the change is not adverse and does not require prior NRC approval because the NRC has already endorsed this methodology in the GALL report (NUREG-1801). The UFSAR described design function is not adversely impacted. There is no adverse change to a procedure that controls an SSC design function for this activity, no test or experiment required, and no change to technical specifications or the Facility Operating License.

The new methodology is endorsed by the latest revision of the Generic Aging Lessons Learned (GALL) report (NUREG-1801), and "provides an updated method for establishing reference curves and environmental correction factors for use in evaluating the fatigue life of reactor components exposed to light water reactor coolants and operational experience". NUREG/CR-6909 was developed after Peach Bottom received approval for license renewal, and is an appropriate methodology for the application.

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There were no 10CFR 50.59 Evaluation Reports performed / implemented for Unit 1 during this reporting period.

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End of 10CFR 50.59 Report

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Exelon Nuclear
Peach Bottom Atomic Power Station
Independent Spent Fuel Storage Installation (ISFSI)
Docket No. 72-29

BIENNIAL 10CFR 72.48 REPORT
JANUARY 1, 2015 THROUGH DECEMBER 31, 2016
EVALUATION SUMMARIES

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There were no 10CFR 72.48 Evaluation Reports performed / implemented for the PBAPS ISFSI during this reporting period.

End of 10CFR 72.48 Report

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Exelon Nuclear
Peach Bottom Atomic Power Station
Units 1, 2 and 3
Docket Nos. 50-171, 50-277, and 50-278

COMMITMENT REVISION REPORT
JANUARY 1, 2016 THROUGH DECEMBER 31, 2016
CHANGE SUMMARIES

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Letter Source: **NRC Inspection Report 92-21, dated 9/15/92**

Exelon Tracking No.: **T02355**

Nature of Commitment: **Chemistry will incorporate the participation in the inter-laboratory cross check data program into QA/QC procedures**

Summary of Justification:

The current chemistry QA/QC program ensures that the inter-laboratory cross check program will continue to occur. Upgraded procedure quality and site chemistry practices justify the allowance for not tracking this commitment any longer. This commitment is considered to be historical in nature. The corrective actions taken were effective and the station is in compliance with 10CFR 20 requirements.

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Letter Source: **Letter to NRC dated 7/17/85, Response to NRC Inspection Report 85-15 for Unit 2**

Exelon Tracking No.: **T03181**

Nature of Commitment: **Quality Assurance to review open problem reports and advise operations of reports that could affect plant restart**

Summary of Justification:

Upgrades in the plant startup review program have resulted in substantial improvements in the review of open items. The corrective actions previously taken were effective and the station is in compliance with requirements. The current unit restart procedure requires that the plant operations review committee review any outstanding nuclear safety issue for resolution prior to restart. This commitment is considered as historical and may be deleted from future commitment programmatic tracking since upgraded industry / PBAPS standards have eliminated the need for detailed tracking of this commitment. Upgraded procedure quality and site operating practices justify the allowance for deleting this commitment.

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Letter Source: **Letter to NRC dated 10/14/08, Response to NRC Generic Letter 2008-01**

Exelon Tracking No.: **T047671**

Nature of Commitment: **Enhanced acceptance criteria of gas intrusion into systems including the requirement to perform non-destructive testing**

Summary of Justification:

Upgrades in the procedure program have resulted in substantial improvements in this area. The Technical Specifications were upgraded to include appropriate testing requirements of safety systems to manage gas accumulation. The corrective actions taken were effective and the station is in compliance with Technical Specification requirements. This commitment is considered as superseded by the implementation of Technical Specification amendments 297/300 for Units 2 and 3, respectively and are deleted from future commitment programmatic tracking.

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Letter Source: **Letter to NRC dated 9/9/94, Response to NRC Generic Letter 1994-02**

Exelon Tracking No.: **T03567**

Nature of Commitment: **Implement interim corrective actions for addressing thermal hydraulic instabilities for boiling water reactors**

Summary of Justification:

The long term actions of implementing the Detect and Suppress Solution – Confirmation Density (DSS-CD) / Oscillating Power Range Monitor as part of the Maximum Extended Load Line Limit Analysis Plus have been implemented. The Technical Specifications were upgraded to include this feature that automatically suppresses thermal hydraulic instabilities. This was implemented as part of Technical Specification amendments 305/309 for Units 2 and 3, respectively. The corrective actions taken were effective and the station is in compliance with Technical Specification requirements. This commitment is considered as superseded by Technical Specification amendments 305/309 implementation and are deleted from future commitment programmatic tracking.

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Letter Source: **Letter to NRC dated 7/2/01, License Renewal Submittal**

Exelon Tracking No.: **T04336**

Nature of Commitment: **Closed cooling water monitored parameters, frequency of sampling and acceptance criteria are based on EPRI TR-107396-2008, Closed Cooling Water Chemistry Guidelines**

Summary of Justification:

In 2013, the industry standard for closed cooling water chemistry was upgraded (EPRI 3002000590–2013). PBAPS has implemented this upgraded standard, which is based on more

recent industry experience, knowledge and strategy for closed cooling water chemistry control aging management. These changes provide additional margin to the operational control limits. This change will not produce a negative impact to structures, systems or components that perform a safety function.

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Letter Source: **Letter to NRC dated 11/3/11, License Amendment Request for Spent Fuel Pool Rack Inserts (License Amendments 287/290 for Units 2 and 3, respectively)**

Exelon Tracking No.: **T04785**

Nature of Commitment: **Boraflex® monitoring program will continue to be maintained until installation of rack inserts is completed**

Summary of Justification:

Spent fuel pool rack inserts have been installed, thereby removing the need to credit the previous neutron absorbers. The corrective actions taken were effective and the station is in compliance with license amendments 297/300 for Units 2 and 3, respectively. Therefore, the need for interim monitoring of the previous neutron absorbers is no longer required and the programmatic commitment is not required to be tracked within the commitment tracking program.

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End of Commitment Revision Report

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