

Peach Bottom Atomic Power Station www.exeloncorp.com
1848 Lay Road
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10CFR 50.59, 10CFR 72.48

January 28, 2011

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 Nos. DPR-12, DPR-44 and DPR-56
NRC Docket Nos. 50-171, 50-277, 50-278, and 72-29 (ISFSI)

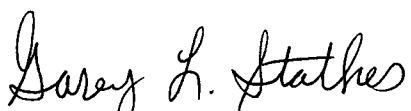
Subject: Biennial 10CFR 50.59 and 10CFR 72.48 Reports for the Period 1/1/2009 through
 12/31/10 and Annual Commitment Revision Report for the Period 1/1/10 through
 12/31/10

Enclosed are the 2009-2010 Biennial 10CFR 50.59 and 10CFR 72.48 Reports and the 2010
Annual Commitment Revision Report as required by 10CFR 50.59 (d)(2), 10CFR 72.48, and
SECY-00-0045 (NEI 99-04). As required to be reported by Off-site Dose Calculation Manual
Specification 3.9.2, there were no major changes to radioactive waste systems at PBAPS during
the reporting period.

There are no 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,



Garey L. Stathes
Plant Manager
Peach Bottom Atomic Power Station

cc: Fred Bower, Senior Resident Inspector, USNRC, PBAPS
 R. R. Janati, Commonwealth of Pennsylvania
 Document Control Desk, USNRC, Washington DC

CCN: 11-01

Attachments

IE47
FSME20
NRC/FSME

**Exelon Nuclear
Peach Bottom Atomic Power Station**

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

2009-2010
Biennial 10CFR 50.59 and 10CFR 72.48 Reports and the 2010 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, 2009 and December 31, 2010. Also, this report identifies commitments that were revised during 2010 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.

**PEACH BOTTOM ATOMIC POWER STATION
UNIT 1, 2 AND 3
DOCKET NOS. 50-171, 50-277, 50-278, 72-29
BIENNIAL 10CFR 50.59 AND 10CFR 72.48 REPORTS AND THE ANNUAL
COMMITMENT REVISION REPORT**

TABLE OF CONTENTS

10CFR 50.59 Report	3
10CFR 72.48 Report	9
Commitment Revision Report	10

**EXELON NUCLEAR
PEACH BOTTOM ATOMIC POWER STATION
UNIT 1, 2 AND 3
DOCKET NOS. 50-171, 50-277, and 50-278**

**BIENNIAL 10CFR 50.59 REPORT
JANUARY 1, 2009 THROUGH DECEMBER 31, 2010
EVALUATION SUMMARIES**

Title: Drywell Equipment Drain Sump Temporary Logic Reconfiguration (ECR 09-00339)

Units Affected: 2

Year Implemented: 2009

Brief Description: The B Drywell Equipment Drain Pump was not operable and the main control room operators were receiving an annunciator on every other pump out. This activity temporarily altered the logic for the two Unit 2 Drywell Equipment Drain Pumps. Instead of having an alternating scheme for the pumps, one pump will perform all the pump outs on the high-high setpoint. Changing the pump setpoint to the high-high level will significantly reduce the number of cycles that the A Drywell Equipment Drain Pump will have to run a pump out until the B Drywell Equipment Drain Pump can be repaired in the refueling outage.

Summary of Evaluation:

This activity removes the pump that is not operating and causes the operating pump to start each time on the high-high setpoint. The Drywell Equipment Drain Pumps and associated logic do not perform or support any safety related function. No safety analyses are impacted by this activity. This change does not interface with the flow monitoring from the equipment drains and does not affect the primary containment isolations for the sump drains. The performance of this activity resulted in temporarily reconfiguring the Unit 2 Drywell Equipment Drain Sump Pump logic. This involved a change to equipment that adversely affects a UFSAR described design function. The pump logic is an UFSAR described design function in UFSAR Section 4.10 as a result of the description of the alternating pump start logic. The 50.59 Evaluation determined that this temporary change does 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 drywell sump pumps. 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.

Title: Application of TRACG04 for OPRM Setpoint Determination (ECR 09-00036)

Units Affected: 2 and 3

Year Implemented: 2009

Brief Description: This activity addresses the use of the GEH advanced multi-purpose NSSS thermal-hydraulic transient code TRACG04P for the purpose of determining the Oscillation Power Range Monitor (OPRM) setpoints for Peach Bottom Atomic Power Station. TRACG04P is a revised version of the NRC approved TRACG02 used previously to determine the OPRM setpoints. The TRACG04P code has not been generically approved by the NRC for OPRM setpoint determinations. Therefore, use of TRACG04P constitutes a change in methodology.

OPRM setpoints are determined for each operating cycle as part of the standard reload licensing process performed in accordance with General Electric Standard Application for Reactor Fuel (GESTAR II) methodology. The cycle specific OPRM setpoints are included in the Core Operating Limits Report (COLR).

Summary of Evaluation:

The TRACG thermal-hydraulic code supports the determination of the second set of setpoints, period based detection algorithm (PBDA) trip setpoints. The TRACG thermal-hydraulic code is used to develop a conservative relationship between the change in fuel bundle critical power ratio (CPR) and the hot bundle oscillation magnitude. This conservative relationship is used to determine the Delta CPR Over Initial MCPR Verses Oscillation Magnitude (DIVOM) curve. The DIVOM curve, in conjunction with the initial maximum critical power ratio (IMCPR) and the hot bundle oscillation magnitude, is used by Global Nuclear Fuels (GNF) to determine the OPRM PBDA setpoints. The algorithms used to detect thermal-hydraulic instability related neutron flux oscillations described in Technical Specification BASES B3.3.1.1 are not impacted by this activity. TRACG04P is only used in the setpoint determination.

The slope of the DIVOM curve represents the thermal-hydraulic responsiveness of the fuel to a given oscillation magnitude, where a steeper slope is more conservative than a flatter slope. A comparison of the TRACG02 code and the TRACG04P code results confirm that the DIVOM slopes developed using TRACG04P are slightly more conservative than those prepared using the NRC-approved TRACG02 version of the code. Therefore, TRACG04P can be used to support the determination of Peach Bottom cycle specific OPRM setpoints without prior NRC approval. The version of TRACG is below the level of detail discussed in the UFSAR and Technical Specifications, therefore a change to the UFSAR and Technical Specification BASES is not necessary. This change is applicable to both Peach Bottom units. The 50.59 Evaluation was processed as part of the Peach Bottom Unit 3 Cycle 18 Reload Fuel Change Package.

Title: Contaminated Auxiliary Steam System (AR 1019819)

Units Affected: 2 & 3

Year Implemented: 2010

Brief Description: The proposed activity is to continue operation of the Auxiliary Steam System while contaminated. The Auxiliary Steam System is a normally nonradioactive system. Per NRC Bulletin No. 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment", actions should be taken to restrict operation of contaminated systems which are considered nonradioactive. However, if it is considered necessary to continue operation of the system as contaminated, an immediate 10CFR50.59 evaluation of the continued operation of the system as a radioactive system must be performed. Since Peach Bottom is proposing to continue operation of the Auxiliary Steam System, a 10CFR50.59 evaluation is required to evaluate whether NRC approval is required for contaminated operation by considering the level of contamination and any potential releases of radioactivity to the environment.

Continued operation of the Auxiliary Steam System is required to provide heating steam to critical components and buildings. This includes freeze protection of

outdoor components and tanks, plant buildings such as the turbine building, reactor buildings, administrative building, pump structure, and other buildings and could ultimately affect various equipment including the accumulator pressure for Hydraulic Control Units (HCUs). Due to below freezing temperatures, this is a required function to ensure safe operation of Units 2 and 3.

Summary of Evaluation:

Operation of the contaminated Auxiliary Steam System creates the potential for offsite release of contaminated water and/or steam. However, all postulated releases have been determined to be within 10CFR 20 and 40CFR 190 limits. Operation with a contaminated Auxiliary Steam System does not change the safety analyses described in the UFSAR and all ODCM limits are maintained. There are no operational changes to the Auxiliary Steam System. However, there will be no additional loads added to the system while it remains contaminated. In addition, sampling at the Auxiliary Boiler Deaerator (00E037) and the Recombiner Building Heating System Condensate Return Tank (00T548) will be conducted to trend the contamination. This will ensure the tritium concentration used in the dose limit calculation for an uncontrolled release remains bounding. While contaminated, release from the Auxiliary Steam System will be monitored prior to release to the environment.

Since the Auxiliary Steam System is not an initiator of any accident nor affect any accident initiators, the frequency of any accident described in the UFSAR does not change when the Auxiliary Steam System is contaminated. Those SSCs important to safety whose malfunctions are described in the UFSAR are also not affected by the Auxiliary Steam System being contaminated, therefore the frequency of their malfunction are not increased. The consequences of an accident are not changing since the Auxiliary Steam System is not utilized to mitigate any accidents described in the UFSAR and being contaminated does not affect those systems relied upon to mitigate accidents. Since the auxiliary steam system being contaminated does not increase the potential dose released for any accident described in the UFSAR, the consequences of a malfunction of an SSC whose malfunctions were previously evaluated in the UFSAR is not increased. An accident of a different type is not created by the auxiliary steam system being contaminated. Operating the system while contaminated does not change how an SSC important to safety previously evaluated in the UFSAR fails. The auxiliary steam system does not directly affect those systems which are responsible for maintaining the integrity of the fission product barriers nor those systems which affect the controlling parameter(s) used to develop the design basis limits. There is no change in methodology as described in the UFSAR which is required to operate the auxiliary steam system while it is contaminated. Based on the 'No' answers to all eight questions, this activity can be performed without prior NRC approval.

Title: 10CFR 72.212 Report Revision of Certification of Compliance (C of C) 1027 Amendment 1 for the TN-68 Spent Fuel Cask (ECR 10-00061)

Units Affected: 2 & 3

Year Implemented: 2010

Brief Description: This activity is to upgrade the PBAPS Independent Spent Fuel Storage Installation (ISFSI) program at PBAPS to add the usage of the Amendment 1 version of the TN-68 Safety Analysis Report (SAR) for newly constructed TN-68 casks. Amendment 1 involves several changes, which include changing the allowable fuel burn-up, minimum cooling times, decay heat, and fuel enrichment. The amendment also included the use of damaged fuel as authorized contents of the cask and to reduce the cask spacing on the storage

pad. This activity addresses those changes made to the previously approved Amendment 0 design. This 10CFR 50.59 Review is performed to satisfy the requirements of 10CFR 72.212.

Summary of Evaluation:

A 50.59 screening has been prepared to support the proposed activity. Screening question 1 has been answered "yes" since the proposed activity does involve an adverse affect on a UFSAR described design function, namely that as a result of the changes to allow higher fuel in the amendment 1 spent fuel cask, there is a higher dose result for normal, off-normal and accident conditions. Since screening question 1 was answered "yes", a 50.59 evaluation has been prepared. The 50.59 evaluation responses conclude that the proposed conversion to using TN-68 amendment 1 casks does not more than minimally increase the potential for accidents or malfunctions of equipment important to safety, does not create any new accidents or malfunctions of equipment important to safety and does not change any methodologies or design basis limits for fission product barriers described in the UFSAR. Changes to the ISFSI program as a result of implementing the TN-68 C of C amendment 1 do not introduce the possibility of a more than minimal change in the consequences of an accident previously evaluated in the UFSAR because the changes are not an initiator of any new malfunctions and no new failure modes are introduced. The NRC-approved TN-68 SAR for the amendment 1 changes to the cask did not result in any new design basis accidents. Although the dose consequences of certain off-normal / accident events (under the jurisdiction of 10CFR 72) have increased due to the higher fuel allowed to be used in the TN-68 cask, these higher dose consequences have already been reviewed and approved using NRC-approved methodologies outlined in the TN-68 SAR amendment 1. Moreover, these increased minimal consequences are approved by the NRC under the assumptions, methodologies, controls and review and approval required by 10CFR 72. There are no affects on DBLFFPs and the dose consequences are within the approved boundaries of the 10CFR 72 requirements and NRC approval of the TN-68 cask amendment 1.

Title: Application of TRACG04P Version 4.2.60.3 for OPRM Setpoint Determination (ECR 09-00520)

Units Affected: 2 & 3

Year Implemented: 2010

Brief Description: This activity addresses the use of the General Electric Hitachi (GEH) advanced, multi-purpose NSSS thermal-hydraulic transient code TRACG04P, Version 4.2.60.3, for the purpose of determining the Oscillation Power Range Monitor (OPRM) setpoints for Peach Bottom Atomic Power Station. OPRM setpoints are determined for each operating cycle as part of the standard reload licensing process performed in accordance with General Electric's Standard Application for Reactor Fuel (GESTAR II) methodology. The cycle specific OPRM setpoints are presented in the Core Operating Limits Report (COLR). Version 4.2.60.3 of TRACG04P is an upgraded version of the NRC approved TRACG02A program originally developed and licensed to determine OPRM setpoints. Version 4.2.60.3 of TRACG04P has not been generically approved by the NRC for OPRM setpoint determination. OPRM system trip functions are described in the UFSAR and the evaluation of OPRM PBDA setpoints is performed as part of the Peach Bottom cycle specific safety analysis process. NEDO-32465-A is cited in Technical Specification 3.3.1.1 by reference. Therefore, use of TRACG04P Version 4.2.60.3

constitutes a change in methodology requiring evaluation in accordance with 10CFR 50.59. Due to similarities between the Peach Bottom Unit 2 and Unit 3 design / licensing bases, this change is applicable to both units.

Summary of Evaluation:

The TRACG02A version of the TRACG thermal-hydraulic code was approved by the NRC and used in the preparation of NEDO-32465-A during the original design and licensing of the GE OPRM system. In 2006 the TRACG code was upgraded to TRACG04 to support coupling with an improved kinetics model resulting from GE's transition to the PANAC11 version of the 3-dimensional core simulator program PANACEA. In 2009 GE implemented a PC-based version of the TRACG04 program, TRACG04P, Version 4.2.57.11. These earlier software upgrades were evaluated under 10CFR 50.59, as documented in 50.59 evaluations PB-2006-001-E and PB-2009-002-E, respectively. This 50.59 Evaluation has been prepared to support upgrading TRACG04P to Version 4.2.60.3. Version 4.2.60.3 implements fixes to several programming deficiencies. This 50.59 evaluation necessarily addresses all software changes implemented subsequent to the version of the program reviewed and approved by the NRC, TRACG02A.

The slope of the DIVOM curve represents the thermal-hydraulic responsiveness of the fuel to a given oscillation magnitude. Thus, a steeper slope is more conservative than a flatter slope (NEDO-32465-A). Benchmarking of the NRC-approved TRACG02A code and the TRACG04P Version 4.2.60.3 code has determined that the DIVOM slope developed using TRACG04P generates a slightly more conservative (steeper) DIVON slope.

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

End of 10CFR 50.59 Report

**EXELON NUCLEAR
PEACH BOTTOM ATOMIC POWER STATION
INDEPENDENT SPENT FUEL STORAGE INSTALLATION
DOCKET NO. 72-29**

BIENNIAL 10CFR 72.48 REPORT

JANUARY 1, 2009 THROUGH DECEMBER 31, 2010

10CFR 72.48 EVALUATION SUMMARIES

There were no 10CFR 72.48 Evaluations performed / implemented during this reporting period.

End of 10CFR 72.48 Report

**EXELON NUCLEAR
PEACH BOTTOM ATOMIC POWER STATION
UNIT 1, 2 AND 3
DOCKET NOS. 50-171, 50-277, and 50-278**

**COMMITMENT REVISION REPORT
JANUARY 1, 2010 THROUGH DECEMBER 31, 2010**

Letter Source: Licensee Event Report (LER) 3-99-02, Safeguard System Inadvertently Disabled

Exelon Tracking No.: T04180

Nature of Commitment: Develop and Implement pre-job brief / work control standards, new security instructions and self-check standards concerning the implementation of security standards.

Summary of Justification:

The corrective actions taken were performed and the station is in compliance with NRC requirements. Standardization of human error prevention tools are now contained in Exelon standard procedures. These standardized programs include self-check, work control standards, pre-job briefs and proper communication standards. These improved standards are now engrained at the site through the Exelon standard procedures. There is no longer a need to track this commitment.

Letter Source: Letter to NRC dated 7/23/86, Response to NRC Inspection Report 50-278/86-09

Exelon Tracking No.: T01941

Nature of Commitment: Develop and implement procedural control to use the plant process computer to generate a rod position map at the completion of withdrawal of specific control rod groups if the Rod Worth Minimizer is out of service. This control rod position map will be compared by the operator to a planned control rod position map attached to the operator's control rod withdrawal sheet.

Summary of Justification:

At the time of this event, the additional control rod position map check was deemed to be required. Since that time, standardized Exelon processes have been put in place to upgrade the use of various human performance tools. This has resulted in substantial improvements in operator performance through training and the use of human performance tools, including self-check and verifications. These advancements have extensive management oversight as well. Therefore, this commitment is considered to be historical in nature. The corrective actions taken were effective and the station is in compliance with NRC requirements. There is no longer a need to track this commitment.

Letter Source: Letter to NRC dated 7/9/85, Response to 1985 Systematic Assessment of Licensee Performance (SALP)

Exelon Tracking No.: T01979

Nature of Commitment: Institute a Fire Protection Review Checklist to be used for plant modifications in order to evaluate the impact to each new modification on the fire barriers.

Summary of Justification:

This is a historical commitment. The corrective actions taken were effective and the station is in compliance with NRC requirements. Improved standards and practices within the modification process ensure proper review of modification impact to the fire protection program. The modification standardized process has substantially improved since 1985. There is no longer a need to track this commitment.

Letter Source: Licensee Event Report (LER) 2-93-02, Condition Prohibited by Technical Specifications involving a Missed Firewatch

Exelon Tracking No.: T03229

Nature of Commitment: Revise the procedure to track fire impairments to require two individuals to review and approve impairments.

Summary of Justification:

This is a historical commitment. The corrective actions taken were effective and the station is in compliance with NRC requirements. Improved standards and practices involving the management of firewatches are in place. Additionally, heightened human performance standards also obviate the need to maintain this commitment. Therefore, there is no longer a need to track this commitment.

Letter Source: Letter to NRC dated 12/17/79 in response to NRC Bulletin 79-26 involving boron loss of BWR control blades

Exelon Tracking No.: T03556

Nature of Commitment: On a monthly basis, track boron depletion of control rod blades such that the depletion limit is not exceeded.

Summary of Justification:

This is a historical commitment. The corrective actions taken were effective and the station is in compliance with NRC requirements. Improved standards and practices within the nuclear fuels process have addressed this issue and are proceduralized within the Exelon standard documents. Upgraded standardized procedures are in place to ensure appropriate management of control rod blades such that the depletion limits are not encroached upon. Standardized Chemistry trending also confirm that there are no adverse conditions with control rod blade boron depletion. Based on the historical nature of this commitment and upgraded standardized practices, there is no longer a need to track this commitment.

End of Commitment Revision Report
