UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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KING OF PRUSSIA, PA 19406-1415

May 12, 2009

Mr. Charles G. Pardee
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer, Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000277/2009002 and 05000278/2009002

Dear Mr. Pardee:

On March 31, 2009, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The enclosed integrated inspection report documents the inspection results, which were discussed on April 17, 2009, with Mr. William Maguire and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, one inspector-identified Severity Level IV and two self-revealing findings of very low safety significance (Green) findings were identified. These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because these findings have been entered into your corrective action program (CAP), the NRC is treating the findings as non-cited violations (NCVs), consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any of the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. NRC, Washington, DC 20555-0001; and the NRC Resident Inspector at the PBAPS. In addition, if you disagree with the characterization of the cross-cutting aspect of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region 1 and the NRC Resident Inspector at PBAPS.

In accordance with Title 10 of the Code of Federal Regulations (CFR) 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Paul G. Krohn, Chief
Reactor Projects Branch 4
Division of Reactor Projects

Docket Nos.: 50-277, 50-278
License Nos.: DPR-44, DPR-56

Enclosures: Inspection Report 05000277/2009002 and 05000278/2009002
w/Attachment: Supplemental Information

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In accordance with Title 10 of the Code of Federal Regulations (CFR) 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos.:	50-277, 50-278
License Nos.:	DPR-44, DPR-56
Report No.:	05000277/2009002 and 05000278/2009002
Licensee:	Exelon Generation Company, LLC
Facility:	Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3
Location:	Delta, Pennsylvania
Dates:	January 1, 2009 through March 31, 2009
Inspectors:	F. Bower, Senior Resident Inspector M. Brown, Resident Inspector R. Fuhrmeister, Senior Project Engineer G. Johnson, Operations Engineer J. Tomlinson, Operations Engineer E. Torres, Project Engineer
Approved by:	Paul G. Krohn, Chief Reactor Projects Branch 4 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000277/2009002, 05000278/2009002; 01/01/2009 – 03/31/2009; Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3; Maintenance Risk Assessments and Emergent Work Control; Operability Evaluations; Permanent Plant Modifications.

The report covered a three-month period of inspection by resident inspectors and announced inspections by two regional operations engineers and regional project engineers. Two self-revealing Green findings and one inspector-identified Severity Level IV finding were identified. Each was dispositioned as a non-cited violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. Cross-cutting aspects associated with findings are determined using IMC 0305, "Operating Reactor Assessment Program," dated January 2009. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. A self-revealing NCV of 10 CFR 50 Appendix B, Criteria V, "Instructions, Procedures and Drawings" was identified when inadequate work instructions resulted in a momentary shorting of a terminal lead during maintenance, which caused an inadvertent Unit 3, primary containment isolation valve (PCIV) signal and entry into a one-hour shutdown Technical Specification (TS) Action Statement on March 3, 2009. Specifically, the work instructions allowed the technicians to lift and manipulate energized leads on a safety-related pressure switch without providing any guidance as to the risk and consequences that inadvertent grounding of those energized leads could cause. Because the risk and consequences were not considered and an inadvertent grounding occurred, a PCIV signal resulted that closed normally open valves on both the containment atmosphere control (CAC) system and the instrument nitrogen system containment penetrations. In addition, both PCIV valves on containment atmosphere dilution (CAD) system were rendered inoperable which required the operators to enter an unplanned one-hour TS Action Statement (3.6.1.3.B) and would have required a plant shutdown within the following 12 hours. Corrective actions included replacing the blown fuse, entering the issue into the CAP, and making a required 60 day verbal report to the NRC.

  The finding is more than minor because it could reasonably be viewed as a precursor to a significant event. Specifically, the failure to assess the risk of inadvertent grounding of energized leads on safety equipment could pose a credible hazard as an initiating event during plant operation. The finding was of very low safety significance because the valves in question failed closed and did not represent an actual open pathway in the physical integrity of reactor containment. This finding has a cross-cutting aspect in the area of human performance (work control) because the licensee's work instructions did not provide appropriate risk insights regarding the risks associated with potential grounding of the energized leads. [H.3(a)] (Section 1R13)

- Severity Level IV. An inspector-identified, Severity Level IV NCV of 10 CFR 50.59 was identified when PBAPS made a safety analyses change that departed from a method of evaluation described in the UFSAR without obtaining prior NRC approval and a license amendment. Specifically, PBAPS used a spent fuel pool criticality analysis methodology that was not previously approved by the NRC, and did not adopt an NRC-approved methodology *en toto* and apply it consistent with applicable terms, conditions, and limitations of that methodology. Corrective actions for this problem included entering the issue into the CAP and making plans to develop a technical evaluation that would demonstrate, using methodologies approved for PBAPS, that adequate margin to criticality exists for the nonconforming condition presented by degraded Boraflex in the SFP storage racks. Additionally, PBAPS submitted a LAR, to use alternative SFP criticality analyses, to the NRC on June 25, 2008.

  This deficiency was evaluated using the traditional enforcement process since it potentially impacts or impedes the NRC's ability to carry out its regulatory mission, in that, PBAPS did not request and receive prior NRC approval for changes in licensed activities. The finding is more than minor and a Severity Level IV violation because it is similar to example D.5 of Supplement I, "Reactor Operations," to the NRC's Enforcement Policy. Specifically, the finding involved a violation of 10 CFR 50.59 that resulted in conditions evaluated as having very low safety significance (i.e., Green) by the SDP. Using the Phase 1 SDP, the inspectors determined that the condition resulting from the violation of 10 CFR 50.59 screened to Green because it could affect the functionality of the fuel barrier (cladding). (Section 1R18.1)

Cornerstone: Initiating Events

- Green. A self-revealing, Green NCV of Unit 3 TS 3.0.4 was identified by the inspectors on January 26, 2009, when a half-scram occurred on Unit 3, shortly after Unit 3 entered Mode 2 for plant startup. Specifically, the 'A' Wide-Range Neutron Monitoring (WRNM) was inoperable as a result of inadequate procedural guidance regarding adjustments made to the mean square voltage (MSV) offset during the outage (prior to the January 26, 2009, startup). The inadequate procedural guidance allowed adjustments to be made which resulted in the WRNM not making a smooth transition from the counting region to the MSV region of operation, causing the 'A' WRNM to be inoperable and resulting in an unexpected half-scram when the WRNM transitioned from the counting region to the MSV region of operation. As a result, TS 3.3.1.1 requirements for the number of available channels of WRNM short period RPS trip in Mode 2 had not been met. TS 3.0.4 requires that when a LCO is not met, entry into a mode or other specified condition shall only be made when the associated actions to be entered permit continued operation in the mode or other condition specified for an unlimited period of time. Corrective actions included entering the issue into the CAP, conducting an event review, and submitting a License Event Report (LER) to the NRC, and revising the WRNM adjustment procedure.

  The finding is more than minor because it is associated with the procedure quality attribute and adversely affected the Initiating Events Cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions. The finding was of very low safety significance because it did not

contribute to the likelihood that both a reactor trip would occur and that mitigation equipment would not be available. This finding has a cross-cutting aspect in the area of human performance (resources) because the licensee's procedure did not provide adequate guidance to prevent adjusting the MSV offset to an unacceptable value. [IMC 0305 aspect: H.2(c)] (Section 1R15)

Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 2 began the inspection period at 100 percent rated thermal power (RTP). On January 18, power was reduced to 96 percent in response to a trip of the 2 'C' main circulating water pump. The unit was returned to full RTP later the same day. On February 9, the unit was reduced to 84 percent for planned control rod sequence exchange, control rod testing, and other planned maintenance and testing. On February 13, power was reduced to 50 percent for planned water box cleaning, full-core control rod settle testing, and other planned maintenance and testing. On February 15, the unit was returned to full power where it remained until the end of the inspection period, except for brief periods to support planned testing and rod pattern adjustments.

Unit 3 began the inspection period at 100 percent RTP. On January 18, power was reduced to 75 percent RTP in response to an increasing trend of total dissolved combustible gases accumulating in the 3 'C' main power transformer (MPT). On January 20, a unit shutdown was initiated for an unplanned maintenance outage to replace the 3 'C' MPT, the 3 'D' safety/relief valve (SRV), and the 3 'A' reactor recirculation pump shaft seal. On January 29, the unit was returned to full power. On March 6, power was reduced to 96 percent to perform planned control cell friction testing on four control rods and the unit was returned to full power on March 7. On March 12, in response to main turbine bypass valve oscillations, power was reduced in several increments until the unit was stabilized at approximately 85 percent power. On March 15, an unplanned power reduction to 20 percent was conducted and the main turbine was tripped to perform repairs on the electro-hydraulic control system and to perform main condenser waterbox cleaning. The unit was returned to full RTP on March 16, where it remained until the end of the inspection period, except for brief periods to support planned testing and rod pattern adjustments.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 1 Sample)

.1 Impending Adverse Weather Conditions

a. Inspection Scope

On February 11 and 12, the inspectors performed a review of the adequacy of actions PBAPS took to prepare and to respond to the adverse environmental conditions resulting from expected and actual high winds. The inspectors observed plant conditions and verified that operations personnel entered operations procedure (OP) PB-108-111-1001, Revision 3, "Preparation for Severe Weather." The inspectors noted that plant personnel walked down the site to ensure no missile hazards existed. The inspectors also toured selected portions of the plant site to look for loose debris that could become missiles. In accordance with the guidance in the OP, operators removed unnecessary loads from the station blackout power source to enhance its reliability. The inspectors were informed that power was reported to be lost to 14 of 97 emergency sirens, and that high winds were the suspected cause. The inspectors verified with PBAPS's Operations and Emergency Planning staff that the affected counties were

notified that, if needed, route alerting would be required as a compensatory measure while the 14 sirens were inoperable.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q – 4 Samples)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed a partial walkdown of four systems to verify the operability of redundant or diverse trains and components when safety-related equipment was inoperable. The inspectors performed walkdowns to identify any discrepancies that could impact the function of the system and potentially increase risk. The inspectors reviewed applicable operating procedures, walked down system components, and verified that selected breakers, valves, and support equipment were in the correct position to support system operation. The four systems reviewed were:

- Unit 2 Reactor Core Isolation Cooling (RCIC) with Unit 2 High Pressure Coolant Injection (HPCI) Out-of-Service (OOS);
- E-2 Emergency Diesel Generator (EDG) with E-4 EDG OOS;
- 343 Transformer with 2 Startup Transformer OOS; and
- Unit 3 Loop ‘A’ Residual Heat Removal (RHR) with Unit 3 ‘B’ RHR OOS.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q – 5 Samples)

.1 Fire Protection – Tours

a. Inspection Scope

The inspectors reviewed PBAPS’s Fire Protection Plan, Technical Requirements Manual (TRM), and the respective pre-fire action plan procedures to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for the areas examined during this inspection. The fire risk analysis was reviewed to gain risk insights regarding the areas selected for inspection. The inspectors performed walkdowns of five areas to assess the material condition of active and passive fire protection systems and features. The inspection was also performed to verify the adequacy of the control of transient combustible material and ignition sources, the condition of manual firefighting equipment, fire barriers, and the status of any related compensatory measures. The following five fire areas were reviewed for impaired fire protection features:

- Cable Spreading Room and Computer Room, Turbine Building, 150’ Elevation (Fire Zone 78H and 129);

- 2 ‘A’ and 2 ‘B’ Reactor Feed Pump Turbine Lube Oil Reservoir, Turbine Building, 135’ Elevation (Fire Zone 79B);
- Unit 3 Reactor Building North Control Rod Drive Equipment and West Corridor, 135’ Elevation (Fire Zone 13H);
- 3 ‘A’ and 3 ‘C’ RHR Pump and Heat Exchanger (HX) Rooms, Reactor Building, Elevation 91’ 6” and 116’ (Fire Zone 11 and 12A); and
- 2 ‘B’ and 2 ‘D’ RHR Pump and HX Rooms, Reactor Building, Elevation 91’ 6” and 116’ (Fire Zone 3).

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06 – 1 Sample)

.1 Internal Flooding

a. Inspection Scope

The inspectors reviewed selected risk-important plant design features intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed the flood analysis and Updated Final Safety Analysis Report (UFSAR). The inspectors walked down the Unit 3 HPCI pump room to evaluate the condition of penetration seals, watertight doors, and other internal design features to verify that they were as described in the Individual Plant Examination (IPE).

b. Findings

No findings of significance were identified

1R11 Licensed Operator Requalification Program (71111.11Q – 1 Sample)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

On March 17, the inspectors observed two crews of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate. The inspectors also verified that the evaluators were identifying and documenting crew performance problems, and verified that training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance;
- Clarity and formality of communications;
- Ability to take timely actions in the conservative direction;
- Prioritization, interpretation, and verification of annunciator alarms;
- Correct use and implementation of abnormal and emergency procedures;
- Control board manipulations;
- Oversight and direction from supervisors; and

- Ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

Performance was compared to pre-established operator action expectations and successful critical task completion requirements as presented in the following documents:

- OP-AA-101-111, “Roles and Responsibilities of On-Shift Personnel,” Revision 3;
- OP-AA-103-102, “Watchstanding Practices,” Revision 8;
- OP-AA-103-103, “Operation of Plant Equipment,” Revision 0; and
- OP-AA-104-101, “Communications,” Revision 1.

This inspection constitutes one quarterly Licensed Operator Requalification Program sample as defined in Inspection Procedure (IP) 71111.11.

b. Findings

No findings of significance were identified.

.2 Licensed Operator Requalification (71111.11B – 1 Sample)

a. Inspection Scope

The following inspection activities were performed using NUREG 1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 9, and 10 CFR Part 55. The inspectors reviewed documentation of recent operating history found in inspection reports, licensee event reports (LERs), the licensee’s CAP, and the most recent NRC plant issues matrix. The inspectors also reviewed specific events from the licensee’s CAP which indicated possible training deficiencies, to verify that they had been appropriately addressed. The resident inspectors were also consulted for insights regarding licensed operators’ performance.

The operating examinations for the weeks of February 16, February 23, and March 2, were reviewed for quality, performance, and excessive overlap. Three of five written examinations administered in 2008 were similarly reviewed for quality and excessive overlap.

On March 21, the results of the annual operating tests and the written exam for 2009 were reviewed to determine if pass-fail rates were consistent with the guidance of NUREG-1021, Revision 9, “Operator Licensing Examination Standards for Power Reactors” and NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance SDP.” The review verified the following:

- Crew pass rates were greater than 80%. (Pass rate was 100%);
- Individual pass rates on the written examination were greater than 80%. (Pass rate was 100%);
- Individual pass rates on the job performance measures (JPMs) of the operating examination were greater than 80%. (Pass rate was 100%); and
- More than 75% of the individuals passed all portions of the examination. (100% of the individuals passed all portions of the examination).

Observations were made of the dynamic simulator exams and JPMs administered to two crews during the week of March 2. These observations included facility evaluations of crew and individual performance during the dynamic simulator exams and individual performance of JPMs. The remediation plans for a crew and individual's performance during requalification training/evaluations were reviewed to assess the effectiveness of the remedial training.

Simulator performance and fidelity were reviewed for conformance to the reference plant control room. Selected simulator deficiency reports were reviewed to assess licensee prioritization and timeliness of resolution. Simulator testing records were reviewed to verify that scheduled tests were performed.

A sample of records for requalification training attendance, program feedback, reporting, including ten operators medical reports were reviewed for compliance with license conditions, including NRC regulations. Interviews were conducted with a sample of operators to discern their perspectives on simulator fidelity, training effectiveness, and response to feedbacks.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 Samples)

a. Inspection Scope

The inspectors evaluated PBAPS's work practices and follow-up corrective actions for safety structures, systems, and components (SSCs) and identified issues to assess the effectiveness of PBAPS's maintenance activities. The inspectors reviewed the performance history of SSCs and assessed Exelon's extent-of-condition (EOC) determinations for those issues with potential common cause or generic implications to evaluate the adequacy of the PBAPS's corrective actions. The inspectors assessed PBAPS's problem identification and resolution (PI&R) actions for these issues to evaluate whether PBAPS had appropriately monitored, evaluated, and dispositioned the issues in accordance with Exelon procedures, including ER-AA-310, "Implementation of the Maintenance Rule," and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classifications, performance criteria and goals, and Exelon's corrective actions that were taken or planned, to evaluate whether the actions were reasonable and appropriate. The inspectors performed the following three samples:

- Unit 3 WRNM 'A' is Believed to be not Fully Qualified (Work Order (WO) A1697287);
- Indications of Channel Distortion - Peach Bottom Unit 3 (Issue Report (IR) 874398); and
- EOC Review Following the Failure of the Unit 3 HPCI Suction Valve (MO-3-23-57) Failure (IR 895789).

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 7 Samples)

a. Inspection Scope

The inspectors evaluated PBAPS's implementation of their Maintenance Risk Program with respect to the effectiveness of risk assessments performed for maintenance activities that were conducted on SSCs. The inspectors also verified that the licensee managed the risk in accordance with 10 CFR Part 50.65(a)(4) and procedure WC-AA-101, "On-Line Work Control Process." The inspectors evaluated whether PBAPS had taken the necessary steps to plan and control emergent work activities and to manage overall plant risk. The inspectors selectively reviewed PBAPS's use of the online risk monitoring software, and daily work schedules. The activities selected were based on plant maintenance schedules and systems that contributed to risk. The inspectors completed seven evaluations of maintenance activities on the following:

- Emergent Work on Unit 3 when Control Rod 14-55 Drifted in from Full-In (WO A1695471);
- Emergent Work on the Unit 3 Leading Edge Flow Monitor (LEFM) String 'B' Going into Maintenance Mode (WO A1696628);
- Emergent Work on Unit 3 Due to a Spurious 'A' WRNM Short Period Trip (WO 871864);
- Preventive Maintenance on a Pressure Switch Results in a Blown Fuse and an Unexpected Closure of PCIVs (WO 887441);
- Emergent Work on Unit 2 HPCI Due to HPCI Suction Valve from Torus Stopping in Intermediate Position (WO A1702109);
- Emergent Power Reduction Work on Unit 3 for 3 'C' Main Transformer Gassing (IR 868369); and
- Emergent Work on Unit 3 Due to the #1 Main Turbine Bypass Valve Cycling (IR 891763).

b. Findings

Introduction: A self-revealing Green NCV of 10 CFR 50 Appendix B, Criteria V, "Instructions, Procedures and Drawings," was identified by the inspectors when inadequate work instructions resulted in a momentary shorting of a terminal lead during maintenance which caused an inadvertent Unit 3 PCIV signal and entry into a 1-hour shutdown TS on March 3, 2009.

Description: On March 3, two Instrumentation and Control (I&C) technicians were given a WO activity to replace pressure switch (PS)-9087G due to the switch reaching end-of-environmental qualification life. The work activity indicated that the impact to operations would be a loss of position indication and alarm function for PCIV AO-3519 during the duration of this activity, PS-9087G was a TS component and that operations should address the appropriate actions of TS 3.6.1.3 and 3.3.3.1, and that instrument air and nitrogen should remain aligned to AO-3521B to maintain its PCIV operability. The work activity stated that "PS-9087G may be de-energized by lifting leads from PS in J-Box on AO-3-07B-3519. Loss of position indication lights will result during lifting of leads." There was nothing in the work instructions that indicated any potential risks associated with working with energized leads.

The two technicians conducted a pre-job brief prior to beginning work. The brief discussed that they would be working with energized leads and the need to be careful in general, however, it did not consider the specific impacts of grounding an energized lead and blowing a fuse on this system.

Proper verification techniques were used and documented the wires to be lifted. The wires were then lifted from the local pressure switch terminals and taped. While removing the 2nd wire from the pressure switch housing a spark occurred. The technicians immediately notified the Unit 3 reactor operator (RO) and reported that a spark had occurred. The RO verified that the position indication on the valve they were working was still out which was expected. The RO also reported no other issues as he examined the main control room (MCR) panels and there were no lost indications at that time. The RO then granted permission to continue the work. Neither the shift supervisor or the I&C supervisor were notified that the spark had occurred. Both I&C technicians and the RO should have notified supervision based on an unexpected result.

The pressure switch was replaced and the wires were re-landed. The technicians checked voltage across the terminal and found no voltage. During this time, the Unit 3 RO reported they had lost indication to nine PCIVs which included the PCIV being worked. In addition, two of the nine PCIVs had repositioned from opened to closed.

When the RO noticed that indications were unexpectedly lost, the work was immediately stopped and both the I&C supervisor and shift supervisor were notified. Prints M-1S-23, sheet 50 and 51 were reviewed. Fuse 16A-F20 was suspected to be blown. The fuse was checked and verified to be blown. The fuse was replaced and indication was restored to all the affected PCIVs.

The valves that repositioned were the nitrogen compressor outboard PCIV and the inboard primary containment vent valve. The repositioning of these valves was as designed given the blown fuse. In addition, power was lost to both the inboard and outboard drywell exhaust vent isolation to standby gas treatment isolation valves (AO-3509 and AO-3510), rendering both of these PCIVs inoperable on the same penetration.

This event resulted in inoperability of multiple PCIVs. Unit 3 entered multiple limiting condition of operation (LCOs) action statements for inoperable PCIVs, which were exited when the fuse was replaced and operability was restored. Corrective actions included replacing the blown fuse, restoring normal system configurations, and entering the issue into the CAP (IR 887441). This event required a 60-day verbal notification to the NRC based on the event being an invalid engineered safety feature (ESF) actuation caused by an error during a maintenance activity.

Analysis: The inspectors concluded that the performance deficiency was having energized leads lifted on safety-related equipment without having documented work instructions that provided guidance to consider the risk and consequences that inadvertent grounding of those leads would have. The finding is more than minor because it could reasonably be viewed as a precursor to a more significant event. Specifically, the failure to assess the risk caused by the inadvertent grounding of

energized leads on safety equipment during maintenance activities creates the potential for an initiating event during plant operation.

Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," the finding affected the Barrier Integrity cornerstone and was of very low safety significance because the valves in question failed closed and did not represent an actual open pathway in the physical integrity of reactor containment, or involve an actual reduction in the defense-in-depth for the atmospheric pressure control or hydrogen control functions of the reactor containment. This finding has a cross-cutting aspect in the area of human performance (work control) because the licensee's work instructions did not provide appropriate risk insights regarding the risks associated with potential grounding of the energized leads. [H.3(a)]

Enforcement: 10 CFR 50 Appendix B, Criteria V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures or drawings appropriate to the circumstances. Contrary to this, on March 3, 2009, energized leads were lifted, removed, reinstalled, and re-landed on a safety-related pressure switch (PS-9087G), without instructions or a procedure that provided guidance to consider the risk and consequences of inadvertent grounding of the energized leads to the pressure switch or associated cable conduit. Because the risk and consequences were not considered and an inadvertent grounding occurred, a PCIV signal resulted that closed normally open valves on both the CAC system and the instrument nitrogen system containment penetrations. In addition, both PCIV valves on the drywell vent system were rendered inoperable which required the operators to enter an unplanned 1-hour TS Action Statement (3.6.1.3.B), and would have required a plant shutdown within the following 12 hours. Because the finding is of very low safety significance and has been entered into PBAPS's CAP (IR 887441), this violation is being treated as a Green NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy, **NCV 05000278/2009002-01, "Inadequate Work Instructions Result in Inadvertent ESF Actuation**."

1R15 Operability Evaluations (71111.15 - 6 Samples)

a. Inspection Scope

The inspectors reviewed six issues to assess the technical adequacy of the operability evaluations, the use and control of compensatory measures, and compliance with the licensing and design bases. Associated adverse condition monitoring plans, engineering technical evaluations, and operational and technical decision making documents were also reviewed. The inspectors verified these processes were performed in accordance with the applicable administrative procedures and were consistent with NRC guidance. Specifically, the inspectors referenced procedure OP-AA-108-115, "Operability Determinations," and NRC IMC Part 9900, "Operability Determinations & Functionality Assessments for Resolutions of Degraded or Nonconforming Conditions Adverse to Quality or Safety." The inspectors also used TS, TRM, UFSAR, and associated Design Basis Documents as references during these reviews. The following degraded equipment issues were reviewed:

- Operability Basis Not Documented for Unit 3 Control Rods (IR 873799);
- Consideration for Channel Distortion Required for Unit 2 (IR 871443);

- Cask Loader Uranium Weights and Enrichments Incorrect (IR 877260);
- Pressure Seal Leaking on RHR Motor-Operated Valve (MOV) MO-3-10-25A (IR 880282);
- 'A' Emergency Service Water (ESW) Booster Pump Tripped During Testing (IR 883424); and
- Inoperable Channel 'A' WRNM Results in a Condition Prohibited By TSs (IR 871864).

b. Findings

Introduction: A self-revealing, Green NCV of Unit 3 TS 3.0.4 was identified by the inspectors on January 26, 2009, when a half-scram occurred on Unit 3, shortly after Unit 3 entered Mode 2 for plant startup. Specifically, the 'A' WRNM was inoperable as a result of inadequate procedural guidance regarding adjustments made to the mean square voltage (MSV) offset during the outage (prior to the January 26, 2009, startup). The inadequate procedural guidance allowed adjustments to be made which resulted in the WRNM not making a smooth transition from the counting region to the MSV region of operation. This caused the 'A' WRNM to be inoperable and resulted in an unexpected half-scram when the WRNM transitioned from the counting region to the MSV region of operation. As a result, TS 3.3.1.1 requirements for the number of available channels of WRNM short period RPS trip in Mode 2 had not been met. TS 3.0.4 requires that when a LCO is not met, entry into a mode or other specified condition shall only be made when the associated actions to be entered permit continued operation in the mode or other condition specified for an unlimited period of time.

Description: On January 21, 2009, while shutting down Unit 3 to replace the 3 'C' main transformer, operations discovered that Channel 'A' WRNM was not responding as expected. The 'A' WRNM was reading twice as high as the other WRNMs. Operations declared the 'A' WRNM inoperable and initiated a WO to repair the WRNM. This WO was given a high priority because the 'A' WRNM would be required for startup since the 'E' WRNM was also inoperable due to a failed detector. Both the 'A' and 'E' WRNMs are on the 'A' reactor protection system (RPS) trip system. Therefore, only two operable WRNMs channels remained operable on the 'A' trip system. In Mode 2, for a reactor start-up, TS 3.3.1.1 for RPS instrumentation requires at least three operable WRNMs on each RPS trip system.

PBAPS's troubleshooting of the 'A' WRNM determined that noise existed on the WRNM channel and was being seen as MSV flux. MSV flux can be likened to intermediate range neutron flux within the reactor core. To compensate for the noise, the MSV offset was adjusted to a value of 8E9. The I&C procedure, IC-11-00395, "Calibration and Alignment for NUMAC WRNM," required that I&C supervision be notified if the MSV offset is adjusted to a value greater than 3E8. I&C supervision was notified in accordance with the procedure. However, no further actions were taken nor were required by the procedure. In a previous engineering evaluation (AR A1632427-01) addressing a similar condition in 2007, the system manager specifically stated that MSV offset cannot be raised higher than 3E8; however, these comments were not addressed by the personnel performing the work. Adjusting the MSV offset to greater than 3E8 affected how the WRNMs transitioned from the counting region (low counts area) to the MSV region (higher counts area). Raising the MSV offset to a high value forced the WRNM to stay in the counting region longer and eliminated a smooth transition from the counting region to the MSV region. When this smooth transition was eliminated, the

WRNM had a sudden shift from the counting region to the MSV region and this sudden shift caused a short period scram signal to be generated.

After the adjustment was made to the MSV offset, the 'A' WRNM was improperly declared operable and a plant startup was allowed to commence. On January 26, 2009, at 9:00 a.m., Unit 3 entered Mode 2 when the mode switch was placed in startup and control rod withdrawal towards criticality was initiated. At 11:42 a.m., the Unit 3 reactor was declared critical. At 12:00 p.m., a short period half-scram was received from the 'A' WRNM when the channel transitioned from the counting region to the MSV region. The 'A' WRNM was again declared inoperable and another WO was generated. At 12:07 p.m., the half-scram was reset. At 9:16 p.m., the 'A' channel was declared operable because engineering determined that the offset had no adverse affect on the operability of the channel in the MSV region. At 2:44 a.m., on January 27, 2009, the mode switch was placed in Run and the WRNMs were no longer required by TSs.

On February 3, 2009, engineering personnel performed a more detailed review of the data collected during startup. PBAPS concluded that with the MSV offset set to such a high value, the 'A' WRNM would not function properly during a startup and the 'A' WRNM was again declared inoperable.

Corrective actions included entering the issue into the CAP, conducting an event review, and submitting a License Event Report (LER) to the NRC, and revising the WRNM adjustment procedure.

Analysis: The inspectors concluded that the performance deficiency was the licensee entering Mode 2 with only two WRNMs being operable on RPS trip system 'A'. TS 3.3.1.1 requires three WRNMs to be operable in Mode 2 on RPS trip system 'A'. TS 3.0.4 prohibited entry into Mode 2 with the requirements of TS 3.3.1.1 not being met. The finding was more than minor because it is associated with the procedure quality attribute and adversely affected the Initiating Events cornerstone objective of limiting events that upset plant stability and challenge critical safety functions. Using the Phase 1 worksheet in Manual Chapter 0609, "Significance Determination Process," the finding was of very low safety significance (Green) since it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would not be available.

This finding has a cross-cutting aspect in the area of Human Performance (resources component), because the PBAPS's procedure did not provide correct and adequate guidance to prevent adjusting the MSV offset to an unacceptable value. [IMC 0305 aspect: H.2(c)]

Enforcement: Unit 3 TS 3.0.4 requires, in part, that when a LCO is not met, entry into a mode or other specified condition shall only be made when the associated actions to be entered permit continued operation in the mode or other condition specified for an unlimited period of time.

Contrary to the above, on January 26, 2009, at 9:00 am, Unit 3 transitioned from Mode 3 to Mode 2 with the requirements of TS 3.3.1.1, "RPS" not met and when the associated ACTIONS of TS 3.3.1.1 did not allow operation in Mode 2 for an unlimited period of time. Specifically, with both the 'A' and 'E' WRNM OOS in Mode 2, TS 3.3.1.1 would require the unit to be in Mode 3 within 12 hours. Because the finding is of very low safety significance (Green) and has been entered into PBAPS's CAP (IR 871864), this finding

is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy, **NCV 05000278/2009002-02, "Inoperable 'A' WRNM Results in a Condition Prohibited by TSs."**

1R18 Permanent Plant Modifications (71111.18 - 1 Sample)

.1 Review of Plant Modifications and Design Change Control

a. Inspection Scope

The inspectors observed selected ongoing and completed work activities to implement a design change that affected both units, while PBAPS Units 2 and 3 were online. The review was conducted to verify that the design bases, licensing bases and the performance capability of the spent fuel pools (SFPs) were not degraded through design changes. The inspectors reviewed PBAPS's engineering change request (ECR) 07-00415-000, "New Analysis for Degraded Spent Fuel Rack Boraflex." The inspectors also reviewed the associated 10 CFR 50.59 screening (PB-2—8-038-S) and 10 CFR 50.59 evaluation PB-2008-002-E documents and related guidance procedures LS-AA-104, "Exelon 50.59 Review Process," and LS-AA-104-1000, "Exelon 50.59 Resource Manual."

b. Findings

Introduction: An inspector-identified, Severity Level IV NCV of 10 CFR 50.59 was identified when PBAPS made a safety analyses change that departed from a method of evaluation described in the UFSAR without obtaining prior NRC approval and a license amendment. Specifically, PBAPS used a SFP criticality analysis methodology that was not previously approved by the NRC, and did not adopt a NRC-approved methodology *en toto* and apply it consistent with applicable terms, conditions, and limitations.

Description: PBAPS has high density spent fuel storage racks that use a neutron absorbing material (Boraflex) to maintain a subcritical fuel array. The SFP storage racks are designed to maintain an effective neutron multiplication factor (K-effective) of less than or equal to 0.95 as required by TS 4.3.1.1.b. To prevent K-effective from exceeding 0.95 under any conditions, the fuel-loading (K-infinity) of the fuel entering the SFP racks must be less than or equal to 1.362 as required by TS 4.3.1.1.a. PBAPS personnel stated that this value was created assuming a maximum Boron -10 (B-10) degradation of 10 percent and up to 10 centimeter random gapping of the Boraflex panels. These values come from an AEA Technology (AEAT) Report dated July 2000, that the PBAPS considers to be the calculation of record. However, the inspectors noted that this report and a predecessor AEAT Report dated November 7, 1996, do not appear to be part of PBAPS's current licensing basis and do not appear to be consistent with the 1986 analysis that is referenced in Section 10.3 of the UFSAR. The inspectors also observed that the AEAT reports have not been reviewed by the NRC staff. IR 671447 documented the B-10 degradation of the Units 2 and 3 SFP storage racks and projected that the 10 percent loss limit would be reached for Unit 2 by March 2008.

PBAPS recognized that due to the ongoing degradation of the Boraflex material, the existing design basis analysis and TS 4.3.1.1.a would become non-conservative. To address that concern, PBAPS submitted a license amendment request (LAR) to change the K-infinity value in TS to 1.318. PBAPS submitted analyses with the LAR that

concluded that, with a K-infinity value of 1.318, the Boraflex areal density could degrade 15.2 percent and K-effective would remain below the TS limit of 0.95. The license amendment was submitted to the NRC on June 25, 2008 (ML081820348[1]). The inspectors noted that the LAR was not submitted before the Boraflex degradation was predicted (March 2008) to exceed the 10 percent limit, but was submitted before the degradation limit was actually exceeded in the fall of 2008. At the end of the inspection period, the review of this LAR was ongoing.

Separately from the LAR, PBAPS recognized the need to address their projection that the SFP Boraflex degradation would exceed the value specified in the AEAT Report. Specifically, the AEAT Report dated July 2000, concluded that the criticality safety margin for the design reference fuel bundle can be satisfied with a model that assumes up to 10 percent average B-10 areal density loss and up to 10 centimeter random gapping of the Boraflex panels. The inspectors noted that the UFSAR specifies that the SFP Boraflex will contain a minimum B-10 areal density of 0.021 grams per centimeter squared (gm/cm^2). UFSAR Section 10.3.4 states that a document entitled "Design Report of High Density Spent Fuel Storage Racks for PECO Energy Company (PECO), formerly Philadelphia Electric Company, Peach Bottom Atomic Power Station Units 2 and 3, Revision 2," dated July 21, 1986, describes the high density SFP storage racks in detail and contains analyses for criticality concerns. The UFSAR also does not recognize the existence of gaps in the Boraflex panels.

PBAPS appropriately recognized that Boraflex degradation beyond UFSAR limits was a degraded and nonconforming condition needed to be evaluated using 10 CFR 50.59. PBAPS developed and approved an ECR 07-00415-000, dated October 20, 2008, to amend the UFSAR and station procedures. The ECR increased the average B-10 areal density loss limit to 15.2 percent by changing the methodology used in the SPF storage racks' design bases criticality analyses. The ECR also documented PBAPS's review and acceptance of analyses conducted by Global Nuclear Fuels (GNF) [ML081820352] and Northeast Technology Corporation (NETCo) [ML081820349] that defined new Boraflex degradation limits and fuel-loading (K-infinity) restrictions needed to maintain the effective neutron multiplication factor (K-effective) within the TS limit. The inspectors noted that these analyses were also submitted in support of the K-infinity change LAR.

As part of the ECR process, PBAPS reviewed the change against the requirements of 10 CFR 50.59. Specifically, PBAPS conducted a 10 CFR 50.59 screening, numbered PB-2-8-038-S, which determined that a 50.59 evaluation was required because the proposed change involved the use of an alternative evaluation methodology from the method referenced in the UFSAR. The alternative methodology would allow the B-10 areal density loss from the SFP rack Boraflex panels to be up to 15.2 percent to maintain K-effective within the TS limit of 0.95.

The inspectors reviewed PBAPS's 10 CFR 50.59 evaluation (PB-2-8-038-E) which also recognized that the activity changed the UFSAR-described methodologies used to analyze spent fuel rack criticality. PBAPS's evaluation stated that to demonstrate that the proposed change activity does not constitute a departure from a method of evaluation described in the FSAR, it must be demonstrated that the methods employed

[1] Accession numbers in the format of ML081820348 are used to locate documents in the NRC's electronic system for managing agency records (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

in the changed PBAPS analysis have been previously approved by the NRC for the same application. The evaluation concluded that based on NRC approval of the planned new methods, and by virtue of the fact that PBAPS would comply with all terms and conditions of the new methodologies (GNF and NETCo) there was no departure from a method of evaluation described in the UFSAR or used in establishing spent fuel rack design bases.

The 50.59 evaluation stated that the GNF and NETCo analyses used tools and techniques that had been reviewed and approved by the NRC. For example, the evaluation stated that the GNF analysis used the MCNP01A 3-dimensional Monte Carlo program and TGBLA06A 2-dimensioinal lattice physics code. The evaluation noted that these analysis tools were used in other GNF analyses that have been approved by the NRC for other licensees. Evaluation, PB-2-8-038-E, stated that the NETCo analysis performed calculations using the KENO V.a Monte Carlo program, the CASMO-4 lattice physics program and the RACKLIFE Boraflex density monitoring and projection program. Similarly, the evaluation noted that these analysis tools were used in other NETCo analyses that have been approved by the NRC. In particular, the evaluation mentioned a NETCo analysis that had been approved for the Indian Point 2 SFP racks. The inspectors noted that, although PBAPS referenced GNF and NETCo analyses that had been approved for use by other licensees, PBAPS used a combination of these methods in a manner that had not been reviewed and approved by the NRC.

To determine the applicable requirements, the inspectors reviewed the rule (10 CFR 50.59). Inspectors also reviewed relevant NRC, industry and licensee (LS-AA-104-1000, Exelon 50.59 Resource Manual) guidance. The inspectors also discussed the 10 CFR 50.59 evaluation with staff in the NRC's Office of Nuclear Reactor Regulation. The inspectors observed that Section 6.2.8 of procedure LS-AA-104-1000 states that when considering the application of a (previously approved) methodology, it is necessary to adopt the methodology *en toto* and apply it consistent with the applicable terms, conditions, and limitations of the methodology. This is consistent with industry guidance Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, which states that the regulation intends that methodology changes be made *en toto*. Mixing attributes of new and existing methodologies is considered a revision to a methodology and must be evaluated as such. The inspectors observed that NEI 96-07 was reviewed by the NRC staff and endorsed in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes,Tests, and Experiments." The inspectors noted that PBAPS's 50.59 evaluation indicated that the change activity being evaluated was based on a blend of the NETCo and GNF analyses. The inspectors observed that neither analysis was used *en toto*. Rather, PBAPS used an output from a portion of the NETCo analysis as a input to the GNF analysis. Further, the inspectors noticed that although the NETCo analysis performed for PBAPS and Indian Point 2 were similar, they were not identical analysis methods and the applicable terms, conditions, and limitations documented in the NRC SER were not consistently applied. Therefore, the inspectors concluded that PBAPS was not in compliance with 10 CFR 50.59 because PBAPS made safety analyses changes that departed from a method of evaluation described in the UFSAR without obtaining prior NRC approval and a license amendment.

Corrective actions for this problem included entering the issue into the CAP and making plans to develop a technical evaluation that would demonstrate, using methodologies approved for PBAPS, that adequate margin to criticality exists for the nonconforming

condition presented by the degraded Boraflex in the SFP storage racks. Additionally, PBAPS submitted a LAR, to use alternative SFP criticality analyses, to the NRC on June 25, 2008 (ML081820348[2]).

The inspectors also verified that the following inspector-identified issues were appropriately placed into the CAP for evaluation and correction:

- In response to NRC Generic Letter (GL) 96-04, PBAPS made a commitment to perform an analysis to determine the design highest reactivity fuel bundle expected to be in the SFPs and perform a UFSAR change per 10 CFR 50.71(e) to reflect this design reference bundle and to reflect the B-10 areal density in the Boraflex panels (IR 864526);
- A 10 CFR 50.71(e) change was not made to UFSAR Section 10.3.4.1.1.2 to reflect that the Boraflex areal density had degraded below the stated minimum of 0.021 gm/cm^2;
- The UFSAR was not updated in accordance with 10 CFR 50.71(e) to reflect analyses submitted and approved in support of a 1993 license amendment (IR 864346);
- ECR 07-00415-000 stated that design documents supporting the current licensing basis of the plant were not in the PBAPS's records management system (IR 904000); and
- Evaluations in ECR 07-00415-000 did not recognize the Boraflex areal density degradation was within the current licensing basis of the SFP storage racks based on aging management programs submitted and approved by the NRC for license renewal (IR 904000).

Analysis: PBAPS made SFP criticality safety analyses changes that departed from a method of evaluation described in the UFSAR without obtaining prior NRC approval and a license amendment. This is a performance deficiency which is contrary to the requirements of 10 CFR 50.59. Because this was a violation of 10 CFR 50.59, it was considered to be a violation which potentially impedes or impacts the regulatory process. Therefore, such violations are characterized using the traditional enforcement process. This change required prior approval from the NRC before its implementation. Comparing this item to the examples in NUREG 1600 Supplement I, "Reactor Operations," this finding is more than minor because NRC approval would have been required.

The inspectors completed a Significance Determination Review using NRC IMC 0609, Attachment 4, Phase 1 – Initial Screening and Characterization of Findings. Using the Phase I Screening worksheet, the inspectors determined that the condition resulting from the violation of 10 CFR 50.59 affected the functionality of the fuel barrier (cladding). Therefore, the issue screens to very low safety significance (Green). Comparing this item to the examples in NUREG 1600 Supplement I, this finding is similar to Item D.5, "Violations of 10 CFR 50.59 that result in conditions evaluated as having very low safety significance (i.e., Green) by the SDP." This is an example of a Severity Level IV violation.

This finding was reviewed for a cross-cutting aspect in accordance with IMC 0612 Section 5.05. It was determined the performance characteristic that was the most

[2] Accession numbers in the format of ML081820348 are used to locate documents in the NRC's electronic system for managing agency records (ADAMS).

significant contributor to the performance deficiency did not align well with the cross-cutting aspects described in the human performance or PI&R component areas. Therefore, no cross-cutting aspect was assigned.

Enforcement: Paragraph (c)(1) of Section 50.59 to Part 50 of Title 10 of the CFR states, in part, that a licensee may make changes in the facility as described in the Final Safety Analysis Report (FSAR) (as updated) without obtaining a license amendment pursuant to Sec. 50.90 only if the change does not meet any of the criteria in paragraph (c)(2). Paragraph (c)(2) states that a licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, if the change, would result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses. The implementing procedure, LS-AA-104-1000, "Exelon 50.59 Resource Manual," Section 6.2.8, requires, in part, that the licensee, determine whether the proposed activity constitutes a departure from a method of evaluation by determining if the activity uses new or different methods of evaluation that are not approved by NRC for the intended application. When considering the application of a methodology, it is necessary to adopt the methodology *en toto* and apply it consistent with applicable terms, conditions, and limitations. Mixing attributes of new and existing methodologies is considered a revision to a methodology and must be evaluated as such.

Contrary to the above, between October 13, 2008, and March 17, 2009, PBAPS had in place a 10 CFR 50.59 evaluation and a UFSAR change (ECR 07-00415-000) that departed from a method of evaluation described in the UFSAR without obtaining prior NRC approval and without obtaining a license amendment pursuant to 10 CFR 50.90. Specifically, PBAPS used a SFP criticality analysis methodology that was not previously approved by the NRC, and did not adopt a NRC-approved methodology *en toto* and apply it consistent with the applicable terms, conditions, and limitations. Rather, UFSAR Section 10.3.4 referenced a July 21, 1986, design report that describes SFP criticality analysis that was previously reviewed and approved by the NRC. Because this was a SLIV violation and was documented in PBAPS's CAP as IR 864431, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy, NUREG-1600: **NCV 05000277/2009002-03; 05000278/2009002-03, "Departure from a Method of Evaluation Without Prior NRC Approval."**

1R19 Post-Maintenance Testing (71111.19 - 7 Samples)

a. Inspection Scope

The inspectors observed selected portions of post-maintenance testing (PMT) activities and reviewed completed test records. The inspectors observed whether the tests were performed in accordance with the approved procedures and assessed the adequacy of the test methodology based on the scope of maintenance work performed. In addition, the inspectors assessed the test acceptance criteria to evaluate whether the test demonstrated that the tested components satisfied the applicable design and licensing bases and the TS requirements. The inspectors reviewed the recorded test data to verify that the acceptance criteria were satisfied. The inspectors reviewed seven PMTs performed in conjunction with the following maintenance activities:

- Replace Digital Control Valve (DCV) SV-3-03A-13120GC on Hydraulic Control Unit (HCU) 46-51 (WO C0226484);

- 2 ‘C’ RHR HX Post-Maintenance Leakage Observed (WO C020077-80);
- 2 ‘C’ Service Water Pump Leak (WO M1701720);
- Unit 2 CAD/CAC Analyzer Failure (WO M1702392);
- Replace E322 Breaker Emergency Control Switch (WO M1702904);
- MO-3-10-013D EOC (WO C0228319); and
- Replace Unit 2 HPCI Pump Discharge Pressure Indicator PI-081 (WO M1704755).

The inspectors verified that issues identified during the PMT were entered into the CAP (IR 888089).

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 - 1 Sample)

.1 Peach Bottom Unit 3 Maintenance Outage

a. Inspection Scope

PBAPS conducted a maintenance outage on Unit 3 from January 21 through January 27, to replace the 3 ‘C’ MPT, the ‘A’ reactor recirculation pump shaft seal, and the ‘D’ SRV. During the outage, the inspectors reviewed the station’s work schedule and the Outage Risk Assessment Management (ORAM) Plan against procedures OU-PB-104, "Shutdown Safety Management Program;” OU-PB-104-1001, "Shutdown Risk Management for Outages;” and OU-AA-103, "Shutdown Safety Management Program." The ORAM plan was reviewed to confirm that PBAPS had appropriately considered risk, industry experience, and previous site specific problems in developing and implementing a plan that maintained shutdown safety defense-in-depth. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored the activities listed below to verify PBAPS controls over the outage activities:

- Observed the control room operators removing the main generator from the grid, completing a soft shutdown of Unit 3, including stabilizing the plant in Mode 3;
- Conducted a walkdown of selected drywell areas to check for unidentified leakage or other discrepant conditions;
- Configuration management, including maintenance of defense-in-depth, commensurate with the outage plan for the key safety functions and compliance with the applicable TS when taking equipment OOS;
- Monitoring of decay heat removal operations;
- Monitoring reactor water inventory controls, including flow paths, configurations, alternative means for inventory additions, and controls to prevent inventory loss;
- Monitoring the status and configuration of electrical systems and switchyard activities to ensure that TS were met;
- Monitored activities that could affect reactivity; and
- Monitored emergent work activities related to the 3 ‘C’ main power transformer.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 7 Samples including 1 IST and 1 Reactor Coolant System (RCS) Leakage)

a. Inspection Scope

The inspectors reviewed and observed selected portions of the following surveillance tests (STs), and compared test data with established acceptance criteria to verify the systems demonstrated the capability of performing the intended safety functions. The inspectors also verified that the systems and components maintained operational readiness, met applicable TS requirements, and were capable of performing design basis functions. The seven STs reviewed and observed included:

- ST-O-003-560-3, Control Rod Exercise - Fully Withdrawn;
- ST-O-052-151-2, E-1 Diesel Generator Simulated Unit 2 Emergency Core Cooling System (ECCS) Signal Auto Start with Offsite Power Available;
- ST-I-010-100-2, RHR Loop 'A' Logic System Functional Test;
- ST-I-063-203-3, Refuel Floor Vent Exhaust Radiation Monitor Calibration and Functional Test for RIS-3-17-458A and C;
- ST-O-052-214-2, E-4 Diesel Generator Slow Start Full Load and Inservice Test (IST);
- ST-O-020-560-2, Reactor Coolant Leakage Test [RCS Leakage]; and
- ST-O-032-301-3, High Pressure Service Water Pump, Valve and Flow Functional and IST [Retest].

The inspectors verified that issues identified during the surveillance testing were entered into the CAP (IR 880239).

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

1EP6 EP Drill Evaluation (71114.06 – 2 Training Samples)

.1 Simulator-Based Training Evolution

a. Inspection Scope

On March 18, the inspectors observed the classification and notification aspects of a licensed operator requalification training examination scenario in the PBAPS simulator. The conduct of the simulator-based training evolution was evaluated in accordance with the guidance in NRC IP 71114.06, "Drill Evaluation." The inspectors verified that training exercise evaluators captured the results for calculation of the Drill and Exercise Performance (DEP) Performance Indicator (PI). The inspectors also verified that weaknesses or deficiencies were captured for the critique of the training exercise. The following simulated events were classified during this one training exercise:

- MS3 – Site Area Emergency, “Failure of Reactor Protection System;” and
- RG1 – General Area Emergency, “Offsite Dose from an Actual of Imminent Release.”

The inspectors reviewed the evaluation, classification, and notification of the observed simulated events to ensure they were accurate and timely or were entered into the CAP (IR 896525) for evaluation and corrective action.

b. Findings

No findings of significance were identified.

.2 Hostile Action Tabletop Training Drill

a. Inspection Scope

On March 24, the inspectors observed the conduct of a hostile action tabletop training drill that postulated an armed assault on the PBAPS site. The drill participants included PBAPS site personnel, Exelon corporate personnel, and personnel representing Federal, State, and local government organizations that would be involved in the response to an actual event. The inspectors’ efforts were focused on the response of personnel in the simulated MCR to the simulated attack. The inspectors also reviewed coordination of response activities between the MCR staff and security supervision. Although the results were not counted for calculating the DEP PI results, the inspectors paid particular attention to the classification of events using the emergency action level (EAL) thresholds and the simulated notification of State, local, and Federal government personnel. The guidance in NRC IP 71114.06, “Drill Evaluation,” was considered during the inspectors’ observation of the drill. The drill was divided into two sessions. Following each session, a facilitated critique was held and each participant group was polled to gather impressions and lessons learned. The inspectors noted that this tabletop drill was conducted to prepare for a larger scale drill to be conducted later in 2009.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

40A2 Identification and Resolution of Problems

.1 Review of Items Entered into the Corrective Action Program

As required by Inspection Procedure 71152, Identification and Resolution of Problems, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed screening of all items entered into the licensee’s corrective action program. This was accomplished by reviewing the description of each new action request/issue report and attending daily management review committee meetings.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153 – 3 Samples)

.1 Unit 3 - Unplanned Downpower in Response to Main Turbine Bypass Valve Oscillations

a. Inspection Scope

On March 12, operators observed unexpected and sporadic oscillations of the number 1 main turbine bypass valve (#1 TBV). As expected, no changes in reactor power or pressure were observed. As an initial precaution, operators reduced reactor power by one half a percent. The inspectors performed MCR observations of key reactor parameters and operator response. Through the day, the #1 TBV oscillations became progressively more frequent and of longer duration. As the #1 TBV percent opening also increased, the #2 TBV began to open. In response, PBAPS further reduced reactor power in increments to 85 percent and the oscillations stopped. The inspectors also monitored PBAPS's response to the event including risk assessments and troubleshooting efforts. PBAPS determined that the repairs should include the replacement of two cards in the turbine electro-hydraulic control (EHC) circuitry. On March 15, power was reduced to 20 percent and the turbine was taken offline to facilitate the cards' replacement. On March 16, the unit was returned to full power. The inspectors observed that the repairs appeared to satisfactorily correct the TBV oscillation issues.

b. Findings

No findings of significance were identified.

.2 (Closed) LER 05000278/2009002-00, Inoperable 'A' WRNM Results in a Condition Prohibited by TSs

A condition prohibited by TS occurred when Unit 3 entered Mode 2 operations for plant startup on January 26, 2009, at 0900 hours. Specifically, the TS 3.0.4 requirements were not met to allow for an entry into a mode of applicability with the 'A' WRNM inoperable. The cause of the inoperable 'A' WRNM was a result of inadequate human performance regarding a technical decision made during the outage (prior to January 26, 2009 startup). The technical decision allowed for entry into Mode 2 after an adjustment was made to the MSV component of the WRNM function resulting in the MSV being inaccurate for a small range of neutron flux while in Mode 2. Individuals involved with the event have been counseled regarding the importance of rigorous technical evaluations when making decisions that could affect TS equipment performance. WRNM adjustment procedures are also being upgraded. There were no actual safety consequences associated with this event. There were no previous similar LERs identified. The licensee documented this event in issue 871864. This LER was reviewed and this violation is being treated as a Green NCV, consistent with Section VI.A of the NRC Enforcement Policy, **NCV 05000278/2009002-02, "Inoperable 'A' WRNM Results in a Condition Prohibited by TSs."** This LER is closed. More information regarding this issue is provided in section IR15 of this report.

.3 HPCI System Torus Suction Valves (1 Sample)

a. Inspection Scope

The inspectors reviewed Exelon's actions to address failures of the Unit 2 and Unit 3 HPCI torus suction valves to stroke fully open during routine testing on March 12 (MO-2-23-058), and March 21 (MO-3-23-057). The inspectors reviewed documents listed in the attachment, observed testing activities in the plant, and discussed the identified problems and evaluation activities with cognizant engineering personnel.

b. Findings

Introduction: The inspectors identified an unresolved item (URI) related to the adequacy of preventive maintenance on MOVs. On March 12 and 21, 2009, HPCI torus suction valves in Unit 2 and Unit 3, respectively, failed to stroke fully open during routine testing. Dry and hardened stem lubricant was identified in both instances. This issue will remain unresolved pending completion of PBAPS's root cause determination and completion of extent of cause and condition evaluations of MOVs in other accident mitigation systems.

Description: On March 12, the Unit 2 HPCI system suppression pool suction valve, MO-2-23-058, failed to fully open when repositioned during quarterly surveillance testing. The valve stroke was interrupted by operation of the motor operator torque switch. On March 21, the Unit 3 HPCI system suppression pool suction valve, MO-3-23-057, failed to fully open when it was repositioned during quarterly testing. The valve stroke was interrupted by actuation of the motor operator torque switch. In both instances, the stem lubricant was found to be dry and hardened. Failures to stroke appeared to be repeat occurrences of a valve failure to stoke event which occurred in October 2007.

PBAPS determined that other safety-related MOVs may be similarly affected by the stem lubricant hardening issue. The EOC and extent of cause evaluations were ongoing at the end of the inspection period. These evaluations included selecting a sample of MOVs to be visually examined for dry and/or hardened stem lubricant. In addition, PBAPS selected a number of MOVs for diagnostic testing with monitoring equipment connected to determine if any degradation of MOV capability had occurred since the last diagnostic testing of that MOV. At the end of the inspection period, these activities were still in progress; therefore, this item remains unresolved: **URI 05000277, 278/2009002-04, "High Pressure Coolant Injection (HPCI) System Torus Suction Valve Failures."**

4OA5 Other Activities

Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status reviews and inspection activities.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 17, 2009, the resident inspectors presented the inspection results to Mr. W. Maguire and other PBAPS staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Exelon Generation Company Personnel

W. Maguire, Site Vice President
G. Stathes, Plant Manager
J. Armstrong, Regulatory Assurance Manager
E. Flick, Engineering Director
L. Bunner, Work Management Director
L. Lucas, Chemistry Manager
R. Franssen, Operations Director
R. Holmes, Radiation Protection Manager
D. DeBoer, Acting Security Manager
T. Wasong, Training Director

NRC Personnel

F. Bower, Senior Resident Inspector
M. Brown, Resident Inspector
R. Fuhrmeister, Senior Project Engineer
G. Johnson, Operations Engineer
J. Tomlinson, Operations Engineer
E. Torres, Project Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000277, 278/2009002-04	URI	HPCI System Torus Suction Valve Failures (Section 4OA3.3)

Opened/Closed

05000278/2009002-01	NCV	Inadequate Work Instructions Result in Inadvertent ESF Actuation (Section 1R13)
05000278/2009002-02	NCV	Inoperable ‘A’ WRNM Results in a Condition Prohibited by TSs (Section 1R15)
05000277, 278/2009002-03	NCV	Departure from a Method of Evaluation without Prior NRC Approval (Section 1R18.1)

Closed

05000278/2009002-00	LER	Inoperable 'A' WRNM Results in a Condition Prohibited by TSs (Section 4OA3.2)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

OP-PB-108-111-1001, Revision 3, Preparation for Severe Weather
WC-AA-101, Revision 15, On-Line Work Control Process
IR 879995, 14 of 97 Sirens Lost Power Due to High Winds 2/12/2009
IR 879846, Severe Weather Walk Down Concerns (North Sub)
IR 879860, Severe Weather Walk Down Concerns (Cooling Towers)
IR 879872, Severe Weather Walk Down Concerns (Screen Structures)
IR 879896, Severe Weather Walk Down Concerns (Water Plant)
IR 879904, Severe Weather Walk Down Concerns (General Areas)

Section 1R04: Equipment Alignment

COL 13.1.A-2, Revision 20, RCIC System
COL 13.1.B-2, Revision 2, RCIC System Control Board Lineup
COL 52A.1.A-2, Revision 14, E-2 Diesel Generator Normal Standby
COL 52A.1.A-2, Revision 12, E-2 Diesel Generator Normal Standby Performed on May 9, 2008
COL 51B.1.B, Revision 6, 343 Transformer
COL 10.1.A-3A, Revision RHR System Setup for Automatic Operation Loop A
COL 10.1.B-3, Revision 4, RHR Common Valve Setup for Automatic Operation

Section 1R05: Fire Protection

PF-11, Revision 2, 3A and 3C RHR Pump and HX Rooms, Reactor Building 91' 6" and 116' Elevation (Fire Zone 11 and 12A)
PF-13H, Revision 3, North CRD Equipment and West Corridor, Unit 3 Reactor Building, 135' Elevation (Fire Zone 13H)
PF-79B, Revision 2, 2A and 2B RFPT Lube Oil Reservoir, Turbine Building 135' Elevation (Fire Zone 79B)
PF-78H, Revision 4, Cable Spreading Room and Computer Room, Turbine Building 150' Elevation (Fire Zone 78H and 129)
PF-3, Revision 3, 2B and 2D RHR Pump and HX Rooms, Reactor Building 91' 6" and 116'\ Elevation (Fire Zone 3)

Section 1R06: Flood Protection

DBD P-T-09, Revision 8, Internal Hazards
IPE Section 3.3.8.2.3, “Reactor Building”
RT-M-045-980-2, Revision 4, Water Tight Door Survey
RT-W-020-930-3, Revision 2, Survey for Flood Barriers in Reactor Building Drainage System
AO 20A.1, Revision 11, Temporary Removal and Installation of Flood Barriers in the Reactor Building Drainage System
M-551, Plumbing and Drainage Reactor Building Unit No. 3 Plan at Elevation 91’6”

Section 1R11: Licensed Operator Requalification Program

OP-AA-101-111, Roles and Responsibilities of On-Shift Personnel, Revision 3;
OP-AA-103-102, Watchstanding Practices, Revision 8;
OP-AA-103-103, Operation of Plant Equipment, Revision 0
OP-AA-104-101, Communications, Revision 1
PSEG0739R, Revision 4, Leak in Primary Containment Requiring Emergency Blowdown

Requalification Program Procedures / Documents:

TQ-AA-106.PLORT-LRTP, Licensed Operator Requalification Program, Revision 1
R.E. Peach Bottom 2009 Requalification Examination Sample Plan
ANSI/ANS-3.4-1983, Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants
Training Review Requests: TQ-AA-224-F100

Condition Reports:

00867065
00743747
00657738
00629970
00577381
00581258
00822542

Simulator-Related Documentation:

ANSI/ANS-3.5-1985, Nuclear Power Plant Simulators for Use in Operator Training

Procedures:

TQ-AA-306, Simulator Management, Revision 0
TQ-AA-302-0105, Simulator Testing Report Annual Update Template, Revision 0
TQ-AA-306, SWR Prioritization
TQ-AA-301-0105, Simulator Review Board Cover Sheet, Revision 2

Simulator Deficiency Reports:

SWR 10171, CRD Flow Controller Upgrade
SWR 10968, Making Water in IC-18

Transient Tests:

TMI Event, 11/1/08
D/W Steam Line Rupture, 11/1/08
LOCA/LOOP, 11/1/08
Power Ramp, 10/13/08
Turbine Trip Within Bypass, 11/1/08
Recirc Pump Trip, 11/1/08
Dual Recirc Pump Trip, 11/1/08
MSIV Closure, 10/14/08
Trip of RFPs, 10/14/08
SCRAM, 10/14/08

Steady State and Computer Tests:

Stability/Mass Balance, 10/13/08
Heat Balance, 12/10/07
Real Time, 12/31/08

Malfunction Tests:

Recirc Jet Pump Riser Failure, 10/28/08
Recirc MG Incomplete Start Sequence, 10/27/08
RCIC Flow Controller Auto Circuit Fails High, 9/29/08
Steam Seal Regulator Fails Open, 10/22/08
Main Generator Hydrogen Leak, 10/25/08
4.16kV Bus E42 Fault, 6/12/08
EHC Hydraulic Pump Trip, 9/30/08
125 VDC panel 2PPC Fault, 12/19/08
Loss of Conowingo Pond, 12/19/08
ARI-42 Valve Failed Closed, 10/22/08

Audits and Assessments:

Training and Qualifications Programs / 08-01 through 08-05

Section 1R12: Maintenance Effectiveness

IR 875571, Clearances Required for Unit 3 Control Rods
A1697287, U3 WRNM ‘A’ is Believed to Be Not Fully Qualified
GE Nuclear WRNM Processing Training Manual
IR 869364, 3A WRNM is Not Responding
IR 871864, Spurious ‘A’ WRNM Short Period Trip
A1696058, 3A WRNMMSV Offset Required Adjustment
A1695466, 3A WRNM is Not Responding
LCO 3-TS-09-0019, 3A WRNM is Inoperable Due to Issuing a Spurious Trip During Reactor Startup while 3E WRNM is Also Inoperable
LCO 3-TS-09-0011, Potential 3A WRNM is Inoperable Due to Failure to Respond During the Shutdown
NT-2-07-041A, Instrument Calibration Sheet – NT-2-07-041A
NT-2-07-041B, Instrument Calibration Sheet – NT-2-07-041B
NT-2-07-041C, Instrument Calibration Sheet – NT-2-07-041C

NT-2-07-041D, Instrument Calibration Sheet – NT-2-07-041D
NT-2-07-041E, Instrument Calibration Sheet – NT-2-07-041E
NT-2-07-041F, Instrument Calibration Sheet – NT-2-07-041F
NT-2-07-041G, Instrument Calibration Sheet – NT-2-07-041G
NT-2-07-041H, Instrument Calibration Sheet – NT-2-07-041H
PLOT-5060C, Initial Licensed Operator Training – WRNM
C0227680, Investigate Noise Issue on 3A WRNM (Online)
IC-11-00395, Calibration and Alignment for NUMAC WRNM
C0227450, Investigate and Repair as Required
SI3N-60C-WRNM-A1MX, WRNM Signal to Noise Ratio and Discriminator Check Completed on 10/4/07
SI3N-60C-WRNM-A1MX, WRNM Signal to Noise Ratio and Discriminator Check Completed on 9/26/07
SI3N-60C-WRNM-A1C2, WRNM 'A' Calibration/ Functional Check, Completed 1/22/09
SI3N-60C-WRNM-A1MX, WRNM Signal to Noise Ratio and Discriminator Check Completed on 1/22/09
NT-3-07-041A, Instrument Calibration Sheet – NT-3-07-041A
NT-3-07-041B, Instrument Calibration Sheet – NT-3-07-041B
NT-3-07-041C, Instrument Calibration Sheet – NT-3-07-041C
NT-3-07-041D, Instrument Calibration Sheet – NT-3-07-041D
NT-3-07-041E, Instrument Calibration Sheet – NT-3-07-041E
NT-3-07-041F, Instrument Calibration Sheet – NT-3-07-041F
NT-3-07-041G, Instrument Calibration Sheet – NT-3-07-041G
NT-3-07-041H, Instrument Calibration Sheet – NT-3-07-041H
SI3N-60C-WRNM-A1C2, WRNM 'A' Calibration/ Functional Check, Completed 9/26/07
IR 872809, Control Rod 14-55 Inoperable for Reactor Pressure Below 850 PSIG
IR 872814, Control Rod 18-55 Inoperable for Reactor Pressure Below 850 PSIG
IR 872815, Control Rod 42-55 Inoperable for Reactor Pressure Below 850 PSIG
IR 871573, Potential for Channel Distortion in D-Lattice Plants
IR 873799, Operability Basis Not Documented for Control Rods
IR 895789, U3 HPCI Torus Suction MO-3-23-57, Split Indication
IR 898030, MO-3-10-13D, Stem Nut Wear & Underthrust
IR 892191, MO-2-23-058, Stopped in Intermediate Position
RT-R-003-960-3, Channel/Control Rod Blade Interference Monitoring
ST-R-003-485-3, CRD SCRAM Insertion Timing of Selected Control Rods
WO A1702109, MO-2-23-058 Stopped in Intermediate Position
WO C0228193, MO-2-23-058 Clean and Lube Valve Stem
IR 892191, MO-2-058 Stopped in Intermediate Position
IR 892284, 10 CFR 50.72(B)(3) NRC 8-hour Non-emergency Notification
IR 206905, CMU/Unplanned TSA: HPCI Valve Excessive Stroke Time
NRC Event 44926, HPCI Declared Inoperable Following Failure of Valve to Fully Open During Surveillance Test
WO C0228446, MO-O-48-0502A, EOC
WO C0228319, MO-3-10-013D, EOC
WO A1704815, RHR Loop B Outboard Discharge Valve, MO-2-10-154B
IR 901501, MO-2-10-154B, As-Found Test Underthrust
WO C0228193, MO-2-23-058, Clean and Lube Valve Stem
WO A1704815, MO-2-10-154B Valve Stem / Actuator Stem Grease Inspection

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

A1695471, U3 Control Rod 14-55 Drifted from Full-In
IR 869309, U3 Control Rod 14-55 Drifted from Full-In
ST-R-003-485-3, Revision 20, CRD Scram Insertion Timing of Selected Control Rods, Performed 1/28/09
RT-R-003-960-3, Revision 0, Channel/Control Rod Blade Interference Monitoring Performed 1/28/09
GE/ Hitachi 10CFR Part 21, Updated Surveillance Program for Channel-Control Blade Interference Monitoring, SC08-05, Revision 1, December 17, 2008
Exelon Channel Distortion Overview, January 25, 2009
A1696628, U3 LEFM String 'B' is in Maintenance Mode
ECR – 08-00077, PM-1051, Heat Balance/ Thermal Power Uncertainty Analysis: Maintenance and Fail
IR 873241, Unit 3 LEFM String 'B' is in Maintenance Mode
C0227596, Investigate/ Repair LEFM Maintenance Mode Problem
IR 871864, Spurious 'A' WRNM Short Period Trip
A1697287, Unit 3 WRNM 'A' is Believed to be not Fully Qualified
IR 869364, 3A WRNM is Not Responding
WO A1702109, MO-2-23-058 stopped in Intermediate Position
WO C0228193, MO-2-23-058 Clean and Lube Valve Stem
IR 892191, MO-2-058 stopped in Intermediate Position
IR 892284, 10 CFR 50.72 (B)(3) NRC 8 hour Non-emergency Notification
IR 206905, CMU/Unplanned TSA: HPCI Valve Excessive Stroke Time
IR 868318, 3C Main Transformer Hydrogen & Methane Above Caution Limits
IR 868156, ACMP Threshold Met for the 3C Main Transformer
IR 602925, Abnormal Noise Coming From 3C Main Transformer Cooling Fans
IR 891763, Unit # #1 Bypass Valve Cycling
AR A1701988, Bypass Valve Number 1 Main Steam to C Condenser
WO C0228160, Install Monitoring Equipment
IR 887441, Fuse 16A-F20 was Blown During PS-9087G Replacement
WC-AA-10, Revision 0, Work Management Process Description
HU-AA-1211, Revision 4, Briefings – Pre-job, Heightened Level of Awareness, Infrequent Plant Activity and Post-job Briefings
WO R0455707, Replace Pressure Switch with EQ switch
WC-AA-104, Revision 10, Review and Screening for Production Risk
LS-AA-125-1002, Revision 6, Tap Root Root Cause Tree Basic Cause Concepts
SA-AA-129, Revision 4, Electrical Safety
HU-AA-081-F-11, Revision 0, Functional Area and Cross-Functional Fundamentals
Print – 6280-M-367, Sheet 1, CAC System

Section 1R15: Operability Evaluations

Operational and Technical Decision Making Process ((IR 869309), Unit 3 Fuel Channel Bow Impacting Control Rod Motion, dated January 24, 2009
Update to GEH Surveillance Program for Channel-Control Blade Interference Monitoring, MFN 08-420, December 19, 2008 (ML083570447)
Surveillance Plan for GNF Thick/Thin Channel-Control Blade Interference Monitoring for BWR/2-5 (C/D-Lattice) Plants (GEH/GNF-0000-0013-9020-02, Revision 4, October 2008) (ML083570450)
BAPS Unit 3 Channel Distortion Issue, Operations Briefing Package, Dated January 25, 2009

Channel Distortion Overview Presentation, Peach Bottom 3 Start-up PORC, January 25, 2009
Peach Bottom 3 Cycle 17 Channel Distortion Friction Test Results Evaluation, JMD-EXN HE3-09-012, dated January 25, 2009
Recommended Cell Friction Monitoring Population – Peach Bottom 3 Cycle 17, JMD-EXN-HE3-09-011
Exelon Transmittal of Design Information (TODI) NF0900017, Peach Bottom 3 Cycle 17 Channel Distortion Monitoring Population, dated January 23, 2009
TODI NF0900021, Peach Bottom Cycle 17 Channel Distortion Monitoring Population – 76 cells
RT-R-003-960-3, Channel/Control Rod Blade Interference Monitoring, Revision 0
NF-AB-135-1420, Establishing Channel Distortion Monitoring Populations, Revision 5
Prompt Investigation, Peach Bottom CASKLOADER Database Issue, dated January 26, 2009
TN-68 Generic TSs
Exelon ISFSI Procedure, SF-300, TN-68 Cask Spent Fuel Assemblies Storage, Selection and Document Requirements
Operability Evaluation 09-02, Pressure Seal Leakage on the RHR MO-3-10-25A
IR 384030, MO-3-10-25A Leaking
AR A1698603, Pressure Seal Leakage on MO-3-10-25A, RHR Inner Injection Valve to Recirc Loop A
AR A1442675, RHR Inner Injection Valve to Recirc Loop A, MO-3-10-25A Leaking
IR 883424, ‘A’ ESW Booster Pump Tripped During Testing
Op-Evaluation 09-003, Revision 0, ‘A’ ESW Booster Pump Tripped on Low Suction Pressure
ST-O-033-310-2, ESW Booster and ECW Pump and Valve Functional Inservice Test performed on 2/24/09

<u>IRs</u>

IR 869309, Unit 3 Control Rod 14-55 Drifted from Full-In
IR 869398, Unit 3 Control Rod 18-55 Slow Insert/Failed to Settle at 00
IR 870887, Unit 3 Control Rod 42-55 Slow and Failed to Settle at 00
IR 871244, Unit 3 Control Rod 46-55 Failed Settle Testing
IR 871573, Potential for Channel Distortion in D-Lattice Plants
IR 872148, Uranium Weights in Peach CASKLOADER Appear to Be Low
IR 869364, 3A WRNM is Not Responding
WO C0227450, Investigate and Repair as Required
LER 09-02, Inoperable ‘A’ WRNM Results in a Condition Prohibited by TSs
IR 897605, Question on 3A WRNM Decision – Background of IR 889065
IR 871047, 3A WRNM MSV Offset Required Adjustment
IR 871864, Spurious ‘A’ WRNM Short Period Trip
PLOT5060C, Revision 003, Initial Licensed Operator Training, WRNM
GE Nuclear Energy - Peach Bottom WRNM Training – 7/7/97
Shift Logs – 1/26/09 – 1/27/09
WO A169287, U3 WRNM-A is Believed to be not Fully Qualified
LCO 3-TS-09-0019, 3A WRNM is Inoperable Due to Issuing a Spurious Trip During Reactor Startup While 3E WRNM is also Inoperable
LCO 3-TS-09-0011, Potential 3A WRNM is Inoperable Due to Failure to Respond During the Shutdown
NT-2-07-041A, Instrument Calibration Sheet – NT-2-07-041A
NT-2-07-041B, Instrument Calibration Sheet – NT-2-07-041B

NT-2-07-041C, Instrument Calibration Sheet – NT-2-07-041C
NT-2-07-041D, Instrument Calibration Sheet – NT-2-07-041D
NT-2-07-041E, Instrument Calibration Sheet – NT-2-07-041E
NT-2-07-041F, Instrument Calibration Sheet – NT-2-07-041F
NT-2-07-041G, Instrument Calibration Sheet – NT-2-07-041G
NT-2-07-041H, Instrument Calibration Sheet – NT-2-07-041H
C0227680, Investigate Noise Issue on 3 'A' WRNM (Online)
IC-11-00395, Calibration and Alignment for NUMAC WRNM
SI3N-60C-WRNM-A1MX, WRNM Signal to Noise Ratio and Discriminator Check Completed on 10/4/07
SI3N-60C-WRNM-A1MX, WRNM Signal to Noise Ratio and Discriminator Check Completed on 9/26/07
SI3N-60C-WRNM-A1C2, WRNM 'A' Calibration/ Functional Check, Completed 1/22/09
SI3N-60C-WRNM-A1MX, WRNM Signal to Noise Ratio and Discriminator Check Completed on 1/22/09
NT-3-07-041A, Instrument Calibration Sheet – NT-3-07-041A
NT-3-07-041B, Instrument Calibration Sheet – NT-3-07-041B
NT-3-07-041C, Instrument Calibration Sheet – NT-3-07-041C
NT-3-07-041D, Instrument Calibration Sheet – NT-3-07-041D
NT-3-07-041E, Instrument Calibration Sheet – NT-3-07-041E
NT-3-07-041F, Instrument Calibration Sheet – NT-3-07-041F
NT-3-07-041G, Instrument Calibration Sheet – NT-3-07-041G
NT-3-07-041H, Instrument Calibration Sheet – NT-3-07-041H
SI3N-60C-WRNM-A1C2, WRNM 'A' Calibration/ Functional Check, Completed 9/26/07

Section 1R18: Plant Modifications

IR 862956, SFP Rack LAR Requests for Additional Information (RAIs) May Invalidate the Unit 2 SFP Rack Licensing Basis
IR 831121, NRC RAI Concerning PBAPS K-Infinity TS Change
IR 864431, Potential Non-Compliance with 10 CFR 50.59
IR 896856, PBAPS Response to NRC RAI Concerning K-Infinity
IR 864526, Potential Commitment Deficiency – GNL 96-04 Response Boraflex
IR 904000, ECR 07-00415 Deficiencies

Shea, Joseph W.; USNRC, "Fuel Storage Criticality Criteria, PBAPS, Units 2 and 3 (TAC NOS. M85756 AND M85757)," Letter to George A. Hunger, Jr.; Philadelphia Electric Company, 28 May 1993.

Ruland, William H.; USNRC, "Spent Fuel Pool Criticality Analysis and Neutron Absorbers," Letter to Steve Craft; NEI, November 6, 2008.

Milano, Patrick D.; USNRC, "Indian Point Nuclear Generating, Unit No. 2 – Amendment Re: Credit for Soluble Boron and Burnup in Spent Fuel Pit (TAC No. MB2989)," Letter to Michael R. Kansler, Entergy Nuclear Operations, May 29, 2002.

Dacimo, Fred; Entergy Nuclear Indian Point 2, "LAR (LAR 01-010) for Spent Fuel Storage Pit Rack Criticality Analysis with Soluble Boron Credit," Letter to USNRC, September 20, 2001

Hunger, G. A.; PECO Nuclear, “Response to GL 96-04, “Boraflex Degradation In SFP Storage Racks,” Letter to USNRC, October 25, 1996.

Peach Bottom, Units 2 and 3 – LAR – Revision to TS 4.3.1.1.a Concerning K-infinity (ML081820348)

Report No. NET-264-2 NP, Revision 1, “Criticality Analysis of the Peach Bottom Spent Fuel Racks for GNF2 Fuel with Boraflex Panel Degradation Projected to May 2012 (ML081820349)

Report 0000-0035-7327-SFP, Revision 2, “GNF2 Spent Fuel Storage Rack Criticality Analysis for PBAPS, Units 2 & 3 (ML081820352)

PBAPS, Unit Nos. 2 and 3: LAR to Revise TS 4.3.1.1.A Concerning K-Infinity (TAC Nos. MD9154 and MD9155) (ML082120486 & ML083030294)

Exelon's Response to Request for Additional Information - Revision to TS 4.3.1.1.a Concerning K-Infinity (ML090690813 & ML090690804).

NRC GL 83-11, Supplement 1, Licensee Qualification for Performing Safety Analyses
NRC GL 96-04, Boraflex Degradation in Spent Fuel Storage Racks

Regulatory Guide 1.187, Guidance for Implementation of 10 CFR 50.59, Changes, Tests and Experiments

NEI 96-07, Guidelines for 10 CFR 50.59 Implementation, Revision 1, November 2000

LS-AA-104, Exelon 50.59 Review Process, Revision 6
LS-AA-104-1000, Exelon 50.59 Resource Manual
RT-R-004-990-2/3, Boraflex Surveillance Using the Racklife Program
RT-R-004-995-2/3, Boraflex Surveillance Using the Badger Test Device

PBAPS PORC Meeting No. 08-17 (10/17/2008) Minutes

Guidance and Recommended Procedure for Maintaining and Using RACKLIFE Version 1.10, EPRI, Palo Alto, CA: 2002. 1003413.

Drawing Number: 457-A-VC-2, Criticality Assessment of the Peach Bottom Spent Fuel Ponds with Degraded Boraflex Panels (AEAT/R/NS/0084 Issue 1), July 2000
Drawing Number: 457-A-VC-1, An Assessment of the Possible Effects of Boraflex Degradation on K-effective for the Peach Bottom Storage Pools (AEAT-0791), November 7, 1996

<u>Section 1R19: Post-Maintenance Testing</u>

WO C0226484, Replace DCV SV-3-03A-13120GC on HCU 46-51
RT-O-003-990-3, Control Rod Stroke Speed Performed on 1/26/09 for Rod 46-51
WO C020077, Replace 2 ‘C’ RHR HX Floating Head
AR A1699328, 2 ‘C’ RHR HX Post-Maintenance Leakage Observed
AR A1546765, Testing for 2 ‘C’ RHR HX Leak

IR 882934, 2 ‘C’ RHR HX PM Leakage Observed
IR 890739, During Start of Pump Lube Water Inlet Line was Leaking WO A1701720, Service Water Lube Water Inlet Line Leaking
WO M1701720, Repair Lube Water Line
WO A1702392, U2 CAD/CAC Analyzer XIC-80411B “Error Alarm”
WO A1686619, ‘B’ CAC CAD Analyzer Failure
WO M1702392, Inspect and Replace Source Module for XIC-80411B
AR A1702904, Replace RMS-2-54-152-1601CS, E322 Control Switch
IR 894291, E322 Breaker Failed to Close
IR 894645, Replace RMS-2-54-152-1601CS, E322 Control Switch
IR894904, Electrical Distribution TS Bases (3.8.7) Potential Needs Upgrade
WO M1704755, Replace PI-2-23-081
WO R1119196, HPCI PVF Cooler Functional IST Test
WO A1704755, Different Indications for Same Parameter
IR 899332, Different Indications for Same Parameter

Section 1R20: Refueling and Other Outage Activities

IR 868369, Unit 3 Power Reduction for 3 ‘C’ Main Transformer Gassing
IR 874398, Indications of Channel Distortion – Peach Bottom 3

Section 1R22: Surveillance Testing

IR 880239, Deficiencies Not Documented Properly
ST-O-052-151-2, Revision 9, E-1 Diesel Generator Simulated Unit 2 ECCS Signal Auto Start with Offsite Power Available – Completed January 15, 2009
WO R1074357 – E-1 EDG Sim ECCS Signal Auto Start w/ Offsite Power
ST-O-020-560-3, RCL Test, Revision 14
ST-O-020-560-2, RCL Test, Revision 12
ST-I-063-203-3, Revision 4, Refuel Floor Vent Exhaust Radiation Monitor Calibration and Functional Test for RIS-3-17-458A and C, Completed March 5, 2009
ST-I-010-100-2, Revision 15, RHR Loop ‘A’ Logic System Functional Test, Completed March 11, 2009
ST-O-052-214-2, Revision 22, E4 Diesel Generator Slow Start Full Load and IST Test Performed 3/19/09
IR 896894, 3CP042 (3C HPSW Pump): Differential Pressure Close to “Action Range”
IR 899886, 3C HPSW Pump Low Capacity

Section 1EP6: EP Drill Evaluation

IR 896525, Shift Manager and STA Incorrectly Determined No Release in Progress
Peach Bottom May 5, 2009, Hostile Action Exercise DRAFT Timeline, Revision F, dated March 11, 2009

Section 4OA3: Event Followup

Partial Listing of Documents Reviewed:

Procedures

ER-AA-302-1006, Revision 7, GL 96-05 Program MOV Maintenance and Testing Guidelines
MA-AA-723-301, Revision 4, Periodic Inspection of Limitorque Model SMB/SB/SBD-000 through

5 Motor Operated Valves
ER-AA-302-1003, Revision 5, MOV Margin Analysis and Periodic Verification Test Intervals
ER-AA-302, Revision 5, MOV Program Engineering Procedure
Issue Reports

00895789 00892191 00896914 00895626 00895918 00898030

Drawings

M-1-S-36, Sheet 2, Revision 73, Schematic Diagram High Pressure Coolant Injection System
M-1-S-36, Sheet 2A, Revision 73, Schematic Diagram High Pressure Coolant Injection System
M-1-S-36, Sheet 3, Revision 74, Schematic Diagram High Pressure Coolant Injection System
M-1-S-36, Sheet 5, Revision 74, Schematic Diagram High Pressure Coolant Injection System
M-1-S-36, Sheet 9, Revision 73, Schematic Diagram High Pressure Coolant Injection System

LIST OF ACRONYMS

ADAMS	Agency-wide Documents Access and Management System
AR	Action Requests/Assignment Reports
CAC	Containment Atmosphere Control
CAD	Containment Atmosphere Dilution
CAP	Corrective Action Program
CFR	Code of Federal Regulations
DCV	Digital Control Valve
DEP	Drill Exercise Performance
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECR	Engineer Change Request
EHC	Electro-hydraulic Control
EDG	Emergency Diesel Generator
EOC	Extent-of-Condition
EP	Emergency Preparedness
ESF	Engineered Safety Feature
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
GL	Generic Letter
GNF	Global Nuclear Fuels
HCU	Hydraulic Control Unit
HPCI	High Pressure Coolant Injection
HX	Heat Exchanger
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IPE	Individual Plant Examination
IR	Issue Report
I&C	Instrumentation and Control
IST	Inservice Test
JPMs	Job Performance Measures
LAR	License Amendment Request
LCO	Limiting Condition for Operation

LEFM	Leading Edge Flow Monitor
LERs	License Event Reports
MCR	Main Control Room
MOVs	Motor-Operated Valves
MPT	Main Power Transformer
MSV	Mean Square Voltage
NCV	Non-cited Violation
NETCo	Northeast Technology Corporation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OP	Operations Procedure
OOS	Out-of-Service
ORAM	Outage Risk Assessment Management
PARS	Publicly Available Records
PBAPS	Peach Bottom Atomic Power Station
PCIV	Primary Containment Isolation Valve
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post-maintenance Testing
PS	Pressure Switch
RCIC	Reactor Core Isolation Cooling
RCL	Reactor Coolant Leakage
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RO	Reactor Operator
RPS	Reactor Protection System
RTP	Rated Thermal Power
SDP	Significance Determination Process
SFPs	Spent Fuel Pools
SRV	Safety/Relief Valve
SSCs	Structure, System, and Component
STs	Surveillance Tests
TBV	Turbine Bypass Value
TRM	Technical Requirements Manual
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order
WRNM	Wide-Range Neutron Monitoring