December 17, 1999

Mr. G. Rainey, President
PECO Nuclear
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box 195
Wayne, Pennsylvania 19087-0195

SUBJECT: NRC INTEGRATED INSPECTION REPORT 05000277/99008, 05000278/99008

Dear Mr. Rainey:

On November 8, 1999, the NRC completed an inspection at the Peach Bottom Atomic Power Station. The enclosed report presents the results of that inspection. We concluded that your staff continued to operate both units safely.

We noted that your staff performed well during the Unit 3 refueling outage. We observed good performance by operations and maintenance personnel throughout the outage. We determined that the station implemented effective applied radiological controls for the outage, including exposure reduction efforts, contamination controls, and internal and external exposure controls. Also, we observed that the engineering support of station operation was typically good during the inspection period; however, we noted three examples where engineering support was deficient. Deficiences existed in the presentation by engineering personnel of the initial post scram review for the Unit 2 automatic shutdown and also in the design of a modification to the Unit 3 recirculation system flow instrument. Additionally, two core spray system welds were identified that were not included in the Inservice Inspection (ISI) program plan. We understand that these items have been entered into your corrective action program.

Based on the results of this inspection, the NRC has determined that three Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the NRC Enforcement Policy. The NCVs involved the heatup of the recirculation system in excess of the Technical Specification limit, the failure to adhere to procedural requirements in the performance of ultrasonic testing of safety-related components, and the failure to include two core spray system welds in your ISI program plan.

If you contest these violations or the severity level of the NCVs, 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, United States Nuclear Regulatory Commission, Washington DC 20555-0001 and the NRC Resident Inspector at the Peach Bottom Atomic Power Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

We appreciate your cooperation.

Sincerely,

**Original Signed By:**

Curtis J. Cowgill, Chief
Projects Branch 4
Division of Reactor Projects

Docket Nos.: 05000277, 05000278
License Nos.: DPR-44, DPR-56

Enclosure: NRC Inspection Report No. 05000277/99008, 05000278/99008

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REGION I

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|---|---|
| License Nos. | DPR-44<br>DPR-56 |
| Report Nos. | 99008<br>99008 |
| Docket Nos. | 05000277<br>05000278 |
| Licensee: | PECO Energy Company<br>Correspondence Control Desk<br>P.O. Box 195<br>Wayne, PA 19087-0195 |
| Facility: | Peach Bottom Atomic Power Station Units 2 and 3 |
| Inspection Period: | September 21, 1999 through November 8, 1999 |
| Inspectors: | A. McMurtray, Senior Resident Inspector<br>M. Buckley, Resident Inspector<br>B. Welling, Resident Inspector<br>R. Nimitz, Senior Radiation Specialist<br>L. Cheung, Engineering Inspector<br>G. Smith, Senior Security Specialist<br>P. Frechette, Security Specialist<br>T. Burns, Engineering Inspector<br>K. Kolaczyk, Engineering Inspector<br>M. Gray, Engineering Inspector |
| Approved by: | Curtis J. Cowgill, Chief<br>Projects Branch 4<br>Division of Reactor Projects |

## EXECUTIVE SUMMARY

Peach Bottom Atomic Power Station
NRC Inspection Report 050277/99008, 05000278/99008

This inspection report included aspects of PECO operations; surveillances and maintenance; engineering and technical support; and plant support areas.

<u>Operations</u>:

- Operations personnel generally performed well following the turbine trip and subsequent scram of the Unit 2 reactor on September 30, 1999. The station staff appropriately evaluated the causes of the automatic shutdown and appropriate equipment problems. Overall, the station review committee performed a thorough review of the event. The inspectors noted that some engineering inputs into the process were initially deficient. Station personnel identified and corrected the causes of the turbine trip and other equipment problems that occurred during the event.

- Following the reactor scram on September 30, 1999, a heatup rate of 170°F in 45 minutes occurred in the 2A recirculation loop. The root cause of this event, as presented in the licensee event report, was in error and will be revised to reflect that the unreliable bottom head drain temperature indication prevented starting a recirculation pump. Procedural problems were a contributing factor in this event and PECO was reviewing the procedures for revision. The heatup of the recirculation system in excess of the technical specification limit of ≤100°F/hr is a Severity Level IV violation and is being treated as a Non-Cited Violation (NCV) consistent with Section VII.B.1.a of the NRC Enforcement Policy. (Section O1.3)

- Operations personnel took prompt, pro-active actions in response to a negative trend in procedure adherence by operators during the Unit 3 shutdown. (Section O7.1)

<u>Maintenance</u>:

- Poor work practices by maintenance personnel performing modifications on control room panels in preparation for the Unit 3 refueling outage caused two engineered safety feature actuations. The corrective actions taken for each event were adequate. (Section M1.2)

- Maintenance personnel took proper actions to remove and address the extent of condition after finding foreign material, a one-inch heavy hex nut, lodged in the impeller of the Unit 3 high pressure coolant injection (HPCI) system pump. (Section M2.2)

<u>Engineering</u>:

- The design changes regarding the vital bus under-voltage relay replacement were properly designed and implemented. The affected design basis documents were appropriately updated or identified for future update. The critical characteristic valuation for the new relays and post modification testing were appropriately completed. (Section E1.1)

- The failure to adhere to procedural requirements in the performance of ultrasonic testing of safety-related components was identified by the inspectors and is a violation of NRC requirements. This violation is identified as a non-cited violation (NCV) in accordance with Section VII.B.1.a to the NRC Enforcement Policy. The violation was placed the licensee's corrective action program as PEP I0010349. (Section E2.1)

  The failure to include two core spray system welds in the ISI program plan was an violation of 10 CFR 50.55a(g)(3), "Inservice Inspection Requirements. This violation is identified as a non-cited violation (NCV) in accordance with Section VII.B.1.a to the NRC Enforcement Policy. The violation was placed the licensee's corrective action program as PEP I0010372. (Section E2.1)

- Inservice inspection was performed acceptably and included appropriate ASME program coverage, qualified personnel, approved procedures, proper implementation, acceptable examination documentation and PECO oversight. The inspections performed were thorough and of sufficient extent to determine the integrity of the components inspected. (Section E2.1)

- Errors were identified in the Inservice Inspection (ISI) program manual and some ISI program drawings. Two welds on the "A" and "C" core spray system were not displayed in ISI drawings or the program manual as required by IWA 1310 of the ASME Code. The nature, significance and cause of this omission were unknown at the close of the report period, however the inspectors confirmed the condition was not present for the "B" and "D" core spray system trains. The core spray system examination schedule was in compliance with the Code requirements. Oversight of contractor supplied non-destructive examination activities by PECO was comprehensive. (Section E2.1)

- Engineering personnel identified a reliability problem with the 351 and SBO power lines from Conowingo Dam, the normal and alternate power supply to the Unit 1 load center and technical support center (TSC). Repeated storm damage events caused a loss of power to the TSC, resulting in loss of emergency assessment capability and NRC notifications. The inspectors noted that PECO has action in progress that is designed to improve reliability. (Section E2.3)

- An engineering modification error caused the flow indication for the 3A recirculation loop to be displayed on the wrong indicator. This event was of minimal consequence, but it revealed several personnel performance deficiencies related to insufficient design reviews, an incomplete acceptance test procedure, and non-adherence to engineering department procedures and guidance. (Section E4.1)

Plant Support:

- PECO continued to maintain its radiation protection program. No changes were identified that adversely affected radiation protection program performance. (Section R1.1)

- Overall, PECO implemented an effective ALARA program. PECO met its 1999 outage ALARA goals and implemented good efforts to reduce personnel occupational exposure for work activities to as low as is reasonably achievable. (Section R1.2)

- Applied radiological controls for ongoing work activities were generally well implemented. Overall, PECO implemented an effective radioactive material and contamination control program. PECO implemented effective assessments of ongoing radiological controls activities. The assessments were of good scope and depth and were performance based. (Sections R1.3, R1.4, R7)

- Security and safeguards activities were conducted in a manner that protected public health and safety. Protected area assessment aids, protected area detection aids, and personnel search equipment were well maintained. Security and safeguards procedures were properly implemented. The security force members (SFMs) demonstrated that they had the requisite knowledge necessary to effectively implement their duties. Management support was adequate to ensure effective implementation of the security program. (Sections S1, S2, S3, S4, S5, S6, S7)

TABLE OF CONTENTS

## Report Details

### Summary of Plant Status

PECO operated both units safely over the period of this report.

Unit 2 began this inspection period at 100% power. On September 30, 1999, a reactor automatic shutdown (scram) resulted from a generator lockout condition and turbine trip caused by a ground fault on the direct current (DC) power system. Following troubleshooting and repairs, the unit was restarted on October 4 and reached 100% power on October 8.

Unit 3 began this inspection period in end-of-cycle coastdown at 82% power. On September 29, 1999, operators shutdown the reactor for refueling outage 3R12. During the outage, numerous maintenance and surveillance activities were conducted and several major plant modifications were installed. The modifications included:

- Installation of an end-of-cycle recirculation pump trip system
- Installation of a power range neutron monitoring system
- Upgrade of the main turbine supervisory instrumentation

The reactor was taken critical on October 24 and it reached 100% power on October 31.

## I. Operations

### O1 Conduct of Operations[1]

O1.1 General Comments (71707)

The inspectors noted good operator performance during the Unit 3 shutdown and startup and that reactor operators adhered to procedures. Communications among operators were clear and peer-checking standards were followed.

During the Unit 3 outage, operations supervision maintained good control and oversight of plant activities. Operations shift supervision took appropriate action to minimize occasional operator distractions caused by high numbers of maintenance personnel in the control room.

The inspectors noted several instances when the reactor operators did not document abnormal equipment/system responses or conditions in the Station Unified Log. The most notable omission was the failure of the Unit 2 reactor operator to log the trip of the reactor feedwater pumps when reactor vessel level reached Level 8 (High Level) approximately 12 minutes after the scram on September 30, 1999.

During inspections of the Unit 3 drywell and the area under the reactor vessel prior to PECO's closing out of these areas before startup after the outage, both areas were

[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

found to be very clean. The control rod drive housing support structure was adequately re-installed.

## O1.2 Unit 2 Turbine Trip and Reactor Scram Review

a. Inspection Scope (71707)

On September 30, 1999, the Unit 2 reactor automatically shutdown due to a trip of the turbine at full power. The inspectors verified that the plant was in a stable condition following the scram and that all required systems functioned properly during the event. The inspectors reviewed the scram review check list procedure. The inspectors also evaluated the root cause investigation and corrective actions for this event to ensure that the cause of the scram and any abnormal equipment problems were identified and repaired.

b. Observations and Findings

PECO's investigation determined that the root cause of the turbine trip was the grounding of an inverter which actuated two generator protection relays and tripped the main generator. Root cause and corrective action information was documented in PEP I0010325.

Two equipment anomalies occurred following the scram:

- When the reactor vessel level was recovered after the scram, the feedwater pumps turbines were stopped per the emergency response procedures. The valve position indicators for the 2 'B' and 'C' reactor feedwater pump turbine (RFPT) control valves indicated the valves were partially open due to a valve position offset identified earlier in the day. This condition prevented the 2 'B' and 'C' feedwater pumps from being restarted as the RFPT control valve position indicators are required to indicate fully closed to permit resetting of the feedwater pumps. Prior to restart of the feedwater pumps, Operations personnel issued a new abnormal operating procedure to address the issue of the position indication problem. This procedure, "AO 6D.3-2: Reactor Feed Pump Operation Following a Scram with a Positive Offset in Control Valve Position Indication," provided instructions to control the reactor feed pumps following a scram when a positive offset exists in the feed pump turbine control valve position indication.

- The 'B' reactor recirculation pump motor generator set did not lockout as expected from the load reject due to a failed coil in the actuating relay. Station personnel replaced this coil prior to starting the pump.

Due to the failure to restart the 2 'C' reactor feedwater pump, the Unit 2 reactor operator overfed the reactor vessel with the 2 'A' pump and reached the Level 8 (High Level) setpoint approximately 12 minutes following the scram. This caused the 2 'A' pump to trip. The 2 'C' pump is normally used to control reactor vessel level following a scram due to a better flow control characteristic of the globe valve in the 'C' train of feedwater discharge piping. The inspectors noted that this information was not documented by Operations personnel in the GP-18 "Scram Review Procedure Check List."

On October 2, 1999, engineering personnel presented the Post Scram Review Data and Analysis to the Plant Operations Review Committee (PORC) for approval of Unit 2 restart. The PORC performed a thorough review of the event and identified errors in the engineering presentation concerning the impact the problems with the position indicators for the 2 'B' and 'C' reactor feedwater control valves had on the operation of the feedwater pumps. The position indicators were required to indicate shut to permit resetting of the feedwater pumps. Engineering personnel also erroneously documented that reactor vessel level was restored following the scram without reaching the Level 8 feedwater pump trip.

The inspectors determined that operations personnel generally performed well following the Unit 2 scram. Information contained in the GP-18 review was complete and accurate except for the high level trip of the 2 'A' feedwater pump. The inspectors concluded that the licensee identified and corrected the cause of the reactor scram and the equipment anomalies that occurred following the scram.

c. Conclusions

Operations personnel generally performed well following the turbine trip and subsequent scram of the Unit 2 reactor on September 30, 1999. The station staff appropriately evaluated the causes of the automatic shutdown and appropriate equipment problems. Overall, the station review committee performed a thorough review of the event. The inspectors noted that some engineering inputs into the process were initially deficient. Station personnel identified and corrected the causes of the turbine trip and other equipment problems that occurred during the event.

O1.3 Unit 2 Reactor Coolant System Heatup Greater Than 100°F/hour and (Discussed) Licensee Event Report 50-277/2-99-006

a. Inspection Scope (71707)

The inspectors reviewed operator actions that resulted in exceeding a 100°F/hr heatup rate of the recirculation system after the reactor scram on September 30, 1999. The inspectors also discussed this issue with operators, supervisors, and plant management.

b. Observations and Findings

Following the Unit 2 scram, operations personnel delayed restarting the recirculation pumps due to an unreliable bottom head drain temperature indication which prevented the verification that Technical Specifications (TS) required prerequisites for starting a recirculation pump were met. Additionally, during the cooldown after the scram, the recirculation loops cooled down at a faster rate than the reactor vessel causing a differential temperature between the recirculation loops and reactor vessel which exceeded the maximum allowed for start of a recirculation pump.

During preparations for initiating shutdown cooling to reduce the differential temperature limit between the recirculation loop and reactor vessel, the 2A recirculation pump suction temperature increased from 106°F to 276°F in 45 minutes. The heatup exceeded the TS allowable reactor coolant system (RCS) heatup rate of ≤100°F/hr. The excessive heatup occurred when the operators drained water to the torus from the 2A recirculation loop causing reactor vessel water to enter the loop.

An engineering evaluation of this condition indicated that the reactor recirculation system was still operable and within the design basis. This evaluation was based on previously analyzed thermal transients for the plant. This event will be accounted for in the total number of thermal cycles the licensee tracks for the life of the plant.

The inspectors noted that the shutdown cooling procedure did not provide a caution to alert the operators that the draining operation to the torus would have the potential for rapidly changing the recirculation loop temperature. Operations personnel planned to permanently revise the shutdown procedure and the procedure for initiating shutdown cooling to provide better control of recirculation loop temperature

Through interviews and discussions with operations and plant management, the inspectors determined that the cause of the excessive heatup rate was incorrectly identified in licensee event report (LER) 50-277/2-99-006, Revision 0. The plant management agreed that the LER would be changed to reflect that an unreliable bottom head drain temperature indication prevented starting a recirculation pump. Station management was reviewing how and why this incorrect information was documented in the LER.

The Peach Bottom Updated Final Safety Analysis Report (UFSAR) section 4.3.4 states that the recirculation pump casing's allowable heatup rate is 100°F/hr, the same as the reactor vessel. Peach Bottom Unit 2 Technical Specifications 3.4.9 requires, at all times, reactor coolant system (RCS) heatup rate be maintained within the limits of ≤100°F/hr. During the activities on October 2, 1999, station personnel, while attempting to satisfy requirements to start a recirculation pump, heated the 2A reactor recirculation system 170°F in 45 minutes. Although within the design basis for the system, this resulted in a heatup rate greater than allowed by TS. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV) consistent with Section VII.B.1.a of the NRC Enforcement Policy. Further corrective actions for this violation were on-going and tracked in the licensee's corrective action program as PEP I0010328. **(NCV 50-277/99-08-01)**

c. Conclusions

Following the reactor scram on September 30, 1999, a heatup rate of 170°F in 45 minutes occurred in the 2A recirculation loop. The root cause of this event, as presented in the licensee event report, was in error and will be revised to reflect that the unreliable bottom head drain temperature indication prevented starting a recirculation pump. Procedural problems were a contributing factor in this event and PECO was reviewing the procedures for revision. The heatup of the recirculation system in excess of the technical specification limit of ≤100°F in one hour is a Severity Level IV violation and is being treated as a Non-Cited Violation (NCV) consistent with Section VII.B.1.a of the NRC Enforcement Policy.

O2.2 Safety System Walkdown - Unit 2 High Pressure Coolant Injection System

a. Inspection Scope (71707)

The inspectors performed a detailed walkdown of selected portions of the Unit 2 high pressure coolant injection (HPCI) system. The inspectors also examined system piping and instrumentation diagrams (P&IDs), the Updated Final Safety Analysis Report (UFSAR), and applicable Technical Specifications.

b. Observations and Findings

On October 12 and 13, 1999, inspectors reviewed Unit 2 HPCI system indications and controls in the main control room and performed a field walkdown of major portions of the system. The inspectors found the system properly aligned, consistent with system P&IDs and the UFSAR. The inspectors observed no significant material deficiencies.

The inspectors identified a minor discrepancy between the UFSAR and a P&ID. The inspectors discussed this discrepancy with the system manager, who drafted an engineering change request to correct the discrepancy.

c. Conclusions

The Unit 2 HPCI system was properly aligned for operation. The system manager took appropriate action for a minor, inspector-identified discrepancy between the system P&ID and the UFSAR.

## O7 Quality Assurance in Operations

O7.1 Response to Negative Trend in Procedure Adherence

a. Inspection Scope (71707)

The inspectors reviewed Quality Assurance (QA) surveillance reports and Performance Enhancement Program (PEP) reports that documented a negative trend in procedure adherence by operators. The inspectors also discussed these reports with QA and operations personnel.

b. Observations and Findings

The inspectors noted that QA documented several instances of operator lapses in attention-to-detail or communications that resulted in minor non-adherences to procedures during the September 29, 1999 shutdown of Unit 3. Operations personnel had also documented similar minor problems with procedure adherence during the previous few months. In response to these findings, operations support personnel initiated an investigation under the PEP process. The PEP issue leader met with several operators to understand the causes and contributing factors that led to these events. Operations management planned to develop detailed corrective actions after the investigation was concluded.

c. Conclusions

Operations personnel took prompt, pro-active actions in response to a negative trend in procedure adherence by operators during the Unit 3 shutdown.

## II. Maintenance

### M1 Conduct of Maintenance

M1.1 General Comments

NRC Inspection Procedures 62707 and 61726 were used in the inspection of plant maintenance and surveillance activities. The inspectors observed and reviewed selected portions of the following maintenance and surveillance test activities:

| Maintenance Observations: | | Observed On: |
|---|---|---|
| C0187806 | Hydraulic Control Unit (HCU) Maintenance | October 15, 1999 |
| C0188039 | Rework Actuator - RHR Containment Spray Outboard Isolation Valve | October 18, 1999 |
| C0187806 | HCU Scram Valve Diaphragms | October 18, 1999 |
| C0187203 | Shutdown Cooling Outboard Isolation Valve - Inspection Actuator | October 18, 1999 |
| A1235512 | Main Turbine Bypass/Control Valve Troubleshooting | October 26, 1999 |
| C0188982 | Power Range Neutron Monitoring System | Various |

| Surveillance Observations: | | Observed On: |
|---|---|---|
| RT-N-013-240-3 | RCIC Overspeed Test Using Aux Steam | October 12, 1999 |
| ST-M-010-621-3 | RHR ‘A’ Primary Containment Isolation Valve Leakage Test | October 19, 1999 |
| RT-O-016-270-3 | Instrument N2 Pressure Decay Test | October 20, 1999 |
| ST-O-054-751-3 | E13 LOCA LOOP Test | October 20, 1999 |
| ST-M-01A-403-3 | IST Position Indication and Lift Check of SRV Tailpipe Vacuum Relief Valves | October 21, 1999 |

| | | |
|---|---|---|
| RT-I-003-490-3 | Scram Backup Valves Test | October 22, 1999 |
| ST-O-080-675-3 | Reactor Pressure Vessel (ASME Class 1) Leakage Pressure Test | October 23, 1999 |
| RT-I-003-490-3 | Testing Scram Backup Valves | November 1, 1999 |
| ST-O-052-122-2 | E2 Diesel Generator RHR Pump Reject Test | November 1, 1999 |
| ST-O-02B-510-2 | Reactor Coolant Temperatures | November 1, 1999 |

The work and testing performed during these activities were professional and thorough. Technicians were experienced and knowledgeable of their assigned tasks. The work and testing procedures were present at the job site and were effectively used. Good pre-job briefs were observed prior to the performance of the maintenance and surveillance activities observed.

M1.2 Main Control Room Modification Maintenance Causes Reactor Protection System Actuations and (Closed) Licensee Event Reports 50-278/3-99-004 and 3-99-005

a. Inspection Scope (62707)

The inspectors reviewed two Unit 3 engineered safety feature (ESF) actuations caused by maintenance personnel performing modification work in the main control room. The inspectors held discussions with the cognizant managers and personnel performing the work. The inspectors also reviewed the LERs for these events.

b. Observations and Findings

On September 1, 1999, while installing a switch for a Unit 3 refueling outage recirculation pump trip modification, a contractor technician inadvertently repositioned the 3A reactor protection system (RPS) alternate power supply switch. This resulted in a temporary loss of power to the 3A RPS, causing a half scram and ESF actuation.

On September 20, 1999, while increasing the size of a hole in the reactor control panel to support a Unit 3 refueling outage power range instrumentation modification, a contractor technician drilled into a wire to the Unit 3B reactor manual scram circuit. This caused a blown fuse, a half scram, and the resultant ESF.

In both cases, RPS responded as designed with no significant safety consequences. Timely notifications were made to the NRC for both events. The inspectors reviewed the followup licensee event reports on-site and identified no further concerns.

Corrective actions for both events included a stop work of all control room modifications, a contractor worker stand down during which the PEPs, the root cause, management expectations for pre-job briefs and job performance; highlighting self-verification and component manipulation were discussed. The first event was reviewed during the second stand down. Field engineers reviewed the remaining modification work after both events to identify possible similar occurrences.

The inspector determined through discussions with PECO management, the technicians performing the work, and the contractor manager that, after the second event, the station changed AG-126, Rev. 4, “Oversight of Contractor Activities,” to include a pre-job checklist and better control of contractor work activities. The pre-job checklist became part of the work packages and included possible mispositioning events due to close proximity to other equipment and prescribed minimum distances to wires from drilling or cutting activities. Also, operations personnel provided more formal briefs to supervision and the technicians doing the job.

Prior to resuming work, the people involved with the drilling on the reactor control panel walked down the job in the simulator prior to start of work to identify possible problems. Physical barriers covering the switches were added or the devices were de-energized in the proximity of the drilling activity. These corrective actions taken as a result of the mispositioning event were adequate to prevent a repeat position error and were considered adequate by the inspectors.

The inspectors performed an on-site review of LERs 3-99-004 and 3-99-005 and identified no violation of NRC requirements. PEP I0010281 was initiated by PECO to evaluate any further corrective actions.

c. Conclusions

Poor work practices by maintenance personnel performing modifications on control room panels in preparation for the Unit 3 refueling outage caused two engineered safety feature actuations. The corrective actions taken for each event were adequate.

## M2 Maintenance and Material Condition of Facilities and Equipment

### M2.1 Foreign Material Found in Unit 3 High Pressure Coolant Injection (HPCI) Pump Impeller

a. Inspection Scope (62707)

During observation of maintenance activities on the Unit 3 HPCI system, the inspectors observed maintenance technicians’ response to the discovery of foreign material in the pump impeller. The inspectors also discussed the possible generic implications of this occurrence with the system manager.

b. Observations and Findings

During disassembly of the Unit 3 HPCI system main pump for scheduled maintenance, maintenance personnel found and removed a one-inch heavy hex carbon steel nut lodged in the first stage impeller near the discharge vanes. The engineering evaluation identified no impact on the pump’s long-term ability to perform its safety function.

The system manager reviewed past surveillance tests, noting satisfactory performance of the HPCI system. The system manager stated that vibration data taken during HPCI pump testing over the last 5 years showed consistent performance. A review of the Unit 3 maintenance activities indicated that in 1989, 1991, and 1993, work was performed that could have introduced the nut into the system. The system manager believed, based on maintenance history, procedure enhancements for control of foreign material, and pump performance of the HPCI system on both units, that there is no foreign material in the system at either unit that would affect pump capability.

Although HPCI was considered operable when the scheduled maintenance began, there were vibration differences between the Unit 2 and Unit 3 HPCI systems; Unit 3 had a slightly greater vibration readings than Unit 2. This was particularly apparent for the gear box end of the pump, which is consistent with finding the foreign material. The vibration data after the removal of the nut from the impeller showed improvement and more closely compared to that of the Unit 2 HPCI. Unit 3 was in a refueling outage during this maintenance, therefore, HPCI was not required to be operable.

Maintenance personnel found no foreign material when a small camera was used to inspect the piping to the booster pump and the outlet of the booster pump. The vanes of the booster pump are larger than the main pump so this type of nut could have traveled through the booster pump and into the main pump.

Considering the inspections performed by maintenance personnel, the history of maintenance, and performance over several years of surveillance testing, the inspectors concluded that it is not likely that any foreign material that would affect the pumps performance still remained in the system. Maintenance personnel took proper actions to remove and address the extent of condition after finding foreign material, a one-inch heavy hex nut in the impeller of the Unit 3 HPCI system pump.

c. Conclusions

Maintenance personnel took proper actions to remove and address the extent of condition after finding foreign material, a one-inch heavy hex nut, lodged in the impeller of the Unit 3 high pressure coolant injection (HPCI) system pump.

## III. Engineering

### E1 Conduct of Engineering

E1.1 Plant Design Change Reviews

a. Inspection Scope (37550)

The inspectors reviewed two plant design changes (98-01304 and 98-01358) to determine whether the changes were properly designed, implemented and in compliance with regulatory requirements. The design changes were also reviewed to determine the extent of engineering involvement, quality of design inputs, implementation of safety evaluations, and post-installation testing requirements.

b. Observations and Findings

The two design changes reviewed were for the under-voltage relay replacement for two 4kV vital buses (E-23 and E-43). The relays were being replaced to correct many calibration problems had occurred in the past as documented in PEP I0004530 and NRC Inspection Report 98-09. The licensee performed an appropriate evaluation and conducted multiple tests to demonstrate that the relays would drop-out and pick-up repeatedly within the acceptable tolerance. The design engineer was directly involved with the testing. The licensee completed the appropriate critical characteristic evaluation for the new relays and detailed 10 CFR 50.59 evaluations. The 10 CFR 50.59 evaluations provided sufficient basis to demonstrate that no unreviewed safety questions were involved in the design changes.

The inspectors found that the design inputs in the Updated Final Safety Analysis Report (UFSAR) and the Technical Specifications (TS) associated with these relays were appropriately incorporated into the design process. The affected test procedures and design drawings were appropriately updated. The necessary changes in the UFSAR and the TS were appropriately identified for a future update.

c. Conclusions

The design changes regarding the vital bus under-voltage relay replacement were properly designed and implemented. The affected design basis documents were appropriately updated or identified for future update. The critical characteristic evaluation for the new relays and post modification testing were appropriately completed.

## E2 Engineering Support of Facilities and Equipment

E2.1 Inservice Inspection (ISI)

a. Inspection Scope (73753)

The inspectors reviewed plans and schedules for the current ISI interval (first outage, first period, third interval) to verify compliance with the requirements of American Society of Mechanical Engineers (ASME) Section XI, 1989 edition and 10 CFR 50.55a(g). Areas inspected included ASME Section XI ISI program coverage, weld examination schedules, qualifications and certifications of the nondestructive examination (NDE) personnel, NDE procedures, results of NDE, and oversight of NDE contractor activities. The inspectors observed selected NDE activities, including an ultrasonic (UT) and magnetic particle (MT) examination of a pipe to sweepolet weld on main steam line “A” and the MT examination of the reactor pressure vessel (RPV) head to flange weld. The inspectors reviewed selected portions of the remote video recordings of the visual inspection of the jet pumps. The inspectors also conducted a field walkdown of portions of the core spray system piping to determine if the ISI drawings and program manual listed all the installed welds and attachments. The scope of the walkdown included all piping and attachments from the torus penetration to the core spray pump discharge check valve.

b. Observations and Findings

NDE contractors perform ISI and in-vessel visual inspection (IVVI) examinations and PECO provided oversight that included review and approval of procedures, qualifications and test results. The inspectors reviewed portions of the ultrasonic, penetrant and magnetic particle test procedures used by NDE personnel and found them to be adequate. NDE contractor personnel were site specific trained and qualified in the use of the test procedures. As part of the oversight function, PECO reviewed and approved personnel qualifications, monitored activities, and reviewed test results. The inspectors reviewed the test reports produced for the NDE examinations observed and found them to be in accordance with the NDE procedures and the ASME Code requirements.

PECO had a comprehensive program for the oversight of contracted NDE services. Observations were documented in oversight reports. The inspectors determined the oversight reports provided a useful assessment of ongoing NDE activities.

The inspectors found the inspection implementation consistent with the approved procedures with the exception of the use of procedure GE-UT-106, “Procedure for Manual Ultrasonic Examination of Ferritic Piping Welds.” Ultrasonic examination procedure GE-UT-106 required that the NDE technician perform specific detailed steps to calibrate the ultrasonic system to the required sensitivity levels for flaw detection. The calibration process involved the use of a calibration block which had precision drilled holes and notches machined into the block to represent flaws. The inspector observed that during the calibration of the ultrasonic test system in preparation for the examination of a branch connection weld on the “A” main steam line, the calibration block being used did not have the notches required by the UT procedure. As a result of this inspector-identified finding, the licensee initiated PEP I0010349. Immediate corrective action was taken by conducting a meeting with NDE contract personnel to stress PECO expectations regarding procedural compliance. Additionally, an amendment to the test procedure was immediately initiated to address the fact that the calibration block did not have notches, and provided specific steps for completing the calibration using only the drilled holes. The inspectors performed a technical investigation and determined that the actual test activity was at a level equal to the ASME Code requirement. The inspectors concluded that no increase in risk resulted from the change.

The failure to adhere to the procedural requirements during the performance of NDE on safety related systems, structures and components is a violation of the requirements stated in 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which requires, in part that, “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.” This severity level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. The violation was placed the licensee’s corrective action program as PEP I0010349. **(NCV 50-278/99-08-02)**

The inspectors reviewed portions of the in-vessel inspection of the jet pump welds in the area of the diffuser assembly to the shroud support plate. The PECO ISI contractor had identified linear indications at repair weld locations on pumps 2 and 10. These indications were characterized as circumferential cracks of approximately 2 inches in length. The licensee had scheduled the examination of ten jet pumps this outage but, upon discovery of the indications, proceeded to examine all twenty pumps. No additional indications of this type were identified. The identification of these cracks was documented by the licensee on Action Request A1232348 and was evaluated by the PECO engineering organization. The inspectors reviewed the flaw evaluation results and concluded that the approach taken for the analysis was broad, thorough and, where assumptions were necessary, were conservative and within guidelines provided in the BWR vessel internals program.

During the walkdown of the “A” and “C” core spray system piping, the inspectors identified some errors in some ISI program drawings and the ISI program manual. Two welds were inappropriately excluded from the ISI program plan. One unidentified weld was in a section of core spray system piping located in the torus enclosure on the “C” train. Another unidentified weld was located in a section of piping upstream of the “A” core spray pump in the core spray pump corner room. Both unidentified core spray system welds were located in ASME Code Class 2 piping. PECO entered observations into the PEP I0010372. PECO’ s initial review concluded the two circumferential welds were inadvertently excluded from the program because out-of-date as-built drawings were apparently used to develop the ISI program drawings and program manual. The inspectors reviewed PECO’s corrective actions and determined they were adequate.

The failure to include the two core spray system welds in the ISI program plan was an violation of 10 CFR 50.55a(g)(3), “Inservice Inspection Requirements,” which states that PECO shall meet the requirements set forth in Section XI of the ASME Boiler and Pressure Vessel Code. Article IWA 1310 of ASME Section XI states that “Components identified... for inspection and testing shall be included in the inservice inspection plan. These components include nuclear power plant items such as vessels, ... piping systems, pumps, valves, core support structures, ... including their respective supports.” This severity level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. The violation was placed the licensee’s corrective action program as PEP I0010372. **(NCV 50-278/99-08-03)**

Article IWC 2500, “Examination and Inspection,” Table IWC-2500-1 of Section XI of the ASME Code states that welds selected for ISI examination shall include 7.5 percent, but not less than 28 welds of all carbon or low alloy steel in Class 2 piping. Unit 3 had 961 welds classified as ASME Code Class 2. Of the 961 welds, 81 were scheduled to receive some form of NDE over the ten year interval. This inspection total exceeded the 69 examinations required by the Code. Fourteen core spray system welds were scheduled for examination over the ten year interval. The inspectors verified the welds scheduled for examination were distributed among terminal ends and structural discontinuities that existed in the system. The core spray system weld examination schedule was in compliance with the Code requirements.

c. Conclusions

The failure to adhere to procedural requirements in the performance of ultrasonic testing of safety-related components was identified by the inspectors and is a violation of NRC requirements. This violation is identified as a non-cited violation (NCV) in accordance with Section VII.B.1.a to the NRC Enforcement Policy. The violation was placed the licensee's corrective action program as PEP I0010349.

The failure to include two core spray system welds in the ISI program plan was an violation of 10 CFR 50.55a(g)(3), "Inservice Inspection Requirements. This violation is identified as a non-cited violation (NCV) in accordance with Section VII.B.1.a to the NRC Enforcement Policy. The violation was placed the licensee's corrective action program as PEP I0010372.

Inservice inspection was performed acceptably and included appropriate ASME program coverage, qualified personnel, approved procedures, proper implementation, acceptable examination documentation and PECO oversight. The inspections performed were thorough and of sufficient extent to determine the integrity of the components inspected.

E2.2 Partially Open Main Steam Relief Valve Causes Reactor Cavity Water to Leak to Torus

a. Inspection Scope (71707 & 37551)

The inspectors observed plant personnel response to an event, during the refueling outage, in which a partially open safety relief valve led to leakage of reactor cavity water to the torus. The inspectors also discussed the event with operations, maintenance, and engineering personnel and reviewed follow-up activities.

b. Observations and Findings

On October 20, 1999, while fuel floor personnel were removing main steam line plugs with the reactor cavity flooded, main control room operators discovered that safety relief valve (SRV) 71C was leaking by its seat. Operators received an alarm indicating that an SRV vacuum relief valve was open and an equipment operator found the vacuum relief valve associated with SRV 71C was leaking reactor cavity water into the drywell. Additionally, operators observed the torus level was increasing, while the spent fuel pool skimmer surge tank level was lowering, indicative of reactor cavity water leaking by SRV 71C to the torus.

Operations personnel responded in a timely manner. Since the leakage rate was within the capacity of the make-up flow rate, the level of the fuel pool/reactor cavity did not decrease and no make-up to the reactor cavity was required.

Plant engineering personnel investigated this event and determined that the leakage occurred due to the SRV air plunger/diaphragm assembly failing to return to its relaxed position following surveillance testing on October 13, 1999. The root cause of this failure was still under review at the end of the inspection period, pending disassembly of the replaced SRV. Engineering personnel considered the generic implications of this event and were developing a long-term corrective action plan to prevent recurrence. Engineers noted that the surveillance testing did not include a visual verification that the air plunger returned to the relaxed position, and they were considering a change to include this verification as part of the surveillance test. PECO's investigation was documented in PEP I0010397.

c. Conclusions

Plant engineering personnel performed a thorough investigation of a partially open safety relief valve that allowed reactor cavity water to leak to the torus during the Unit 3 outage. Engineering personnel considered the generic implications of this event and made plans for long-term actions to prevent recurrence.

E2.3 Loss of Power to the Technical Support Center and Reliability of the 351 Power Line

a. Inspection Scope (37551)

The inspectors reviewed the licensee's activities to improve the reliability of power supplied to the technical support center (TSC) following several loss of power events that usually occurred during adverse weather this past summer. Planned improvements to the Conowingo power lines were discussed with the electrical engineering personnel.

b. Observations and Findings

Power was lost to the 351 line on three separate occasions from July to September 1999 due to storm damage. The loss of the 351 line affects the station blackout (SBO) line and results in a loss of power to the technical support center (TSC). The loss of power to the TSC results in a loss of emergency assessment capability and, if greater than an hour, an one hour non-emergency report to the NRC if required.

The inspectors discussed these events with site engineering personnel and determined that PECO considered the Unit 1 load center and 351 line reliability less than acceptable. In response, PECO initiated a York County Reliability Enhancement Plan to address immediate reliability issues for the 351 and 341 (a backup supply to the 351) lines. Work crews have been assigned to work this project for the remaining of 1999, with a maintenance strategy developed for 2000. The inspectors had no additional questions regarding this matter.

c. Conclusions

Engineering personnel identified a reliability problem with the 351 and SBO power lines from Conowingo Dam, the normal and alternate power supply to the Unit 1 load center and technical support center (TSC). Repeated storm damage events caused a loss of power to the TSC, resulting in loss of emergency assessment capability and NRC notifications. The inspectors noted that PECO has action in progress that is designed to improve reliability.

**E4 Engineering Staff Knowledge and Performance**

E4.1 Recirculation Loop Flow Instrumentation Modification

a. Inspection Scope (37551)

The inspectors reviewed issues related to a design error that caused the flow indication for the 3A recirculation loop to be displayed on the wrong indicator. The inspectors discussed this issue with engineering personnel, and reviewed design documents, engineering modification procedures, and corrective action documentation.

b. Observations and Findings

On October 20, 1999, during the Unit 3 outage, operators discovered that the indication for the 3A recirculation loop flow was displayed on the indicator for the 3B loop. Operators stopped the pump, and engineers corrected the problem through an engineering modification. This issue did not affect plant operation since the reactor was shutdown.

The recirculation pump flow instrumentation was modified during the outage under engineering change request (ECR) 98-03047. The flow indication configuration is unusual, in that the loop ‘A’ flow is shown on the ‘92B’ indicator, and the loop ‘B’ flow is displayed on the ‘92A’ indicator. This arrangement existed since plant construction. Engineering personnel recognized this unusual configuration, but did not ensure that it was maintained in the modification. In addition, the design error was not identified during acceptance testing.

Investigation by PECO into the causes of this problem revealed a number of deficiencies in the design and installation of the modification:

- Engineers did not ensure that the configuration of the recirculation flow indicators was fully captured in the design change process.
- Reviews of the modification were insufficient to identify the design error.
- The acceptance test procedure did not adhere to engineering procedures regarding ‘one point beyond testing’ and thus did not identify the error. ‘One point beyond’ involves testing beyond the modified portion of a circuit.
- The responsible engineer did not fully understand the concept of ‘one point beyond testing.’

The inspectors reviewed the investigation (PEP I0010417) and planned corrective actions, and determined that they were adequate. However, some minor deficiencies were identified. First, inspectors noted that the acceptance test procedure did not include a testing matrix or other suitable verification mechanism, contrary to engineering department guidance. This issue was not revealed in the investigation. Secondly, the inspectors noted that the corrective actions did not specifically address the potential that other engineers may not fully understand the concept of ‘one point beyond testing.’

c. Conclusions

An engineering modification error caused the flow indication for the 3A recirculation loop to be displayed on the wrong indicator. This event was of minimal consequence, but it revealed several personnel performance deficiencies related to insufficient design reviews, an incomplete acceptance test procedure, and non-adherence to engineering department procedures and guidance.

## E8 Miscellaneous Engineering Issues

E8.1 (Closed) VIO 50-277/98-09-01 Failure to Translate Design Basis Information Into Procedures for The Reactor Core Isolation Cooling (RCIC) System

There were two examples associated with this violation. The first example involved the calculated stroking time for the torus suction valves, MO-13-39 & 41, not being properly incorporated into the surveillance test procedure (ST-O-013-301-2). The second example involved an arithmetic error (incorrect piping length) in Engineering Calculation 18247-M-001 that had caused a higher maximum torus water temperature (200°F) for the RCIC pump net positive suction head (NPSH) being used in Emergency Operating Procedure EOP T-102, “Primary Containment Control.” The licensee’s preliminary recalculation at that time indicated that this temperature should have been 187°F.

During this inspection, the inspectors reviewed the revised version (Revision 19) of Procedure ST-0-013-301-2, “RCIC Pump Valve, Flow and Unit Cooler Functional and In-service Test,” dated July 3, 1999, and verified that the stroking time limit for the RCIC torus suction valves had been changed to 37.5 seconds (consistent with the calculation). The inspectors also reviewed the revised version of Calculation 18247-M-001 (dated March 31, 1999) and verified that the calculation had been properly updated. The revised calculation showed that the maximum torus water temperature for an adequate RCIC pump NPSH was 182°F (instead of the preliminary recalculated temperature of 187°F). This temperature was still above the torus water design temperature of 140°F (Section 4.7 of UFSAR) and therefore adequate NPSH was always available during operation. Other calculations (18247-ME-0537 and ME-0695) which used inputs from this calculation were also updated. The licensee also extended their corrective actions to cover the high pressure coolant injection (HPCI) system, which had similar discrepancies. The licensee was still in the process of revising EOP T-102, and had completed the preliminary RCIC and HPCI pump NPSH limit curves based on the newly calculated torus water temperature limits. The licensee stated that the revised procedure would be finalized in November 1999. This process was being tracked by Action Request (AR) A1195026. During the interim, the licensee issued a shift update notice (SUN) for the control room operators’ use.

The licensee was conducting a major review of their engineering calculations, in response to their self-assessment findings, to ensure that calculation results were consistent with the design bases. The scope of this review was documented in two PEPs (I0007052 and I0006493) and was extensive, involving hundreds of calculations in multiple engineering disciplines. This ongoing effort in engineering calculation improvement could reasonably serve as corrective action to prevent recurrence of this violation. The inspectors have no additional concerns with this violation.

E8.2 Follow-up Review of Y2K Remediation Actions

The inspectors performed a review of PECO's Y2K remediation actions using Temporary Instruction (TI) 2515/141, Revision 1, "Review of Year 2000 (Y2K) Readiness of Computer Systems at Nuclear Power Plants." The inspectors verified through maintenance observations and review of maintenance and testing documentation that remediation efforts for the following systems were completed:

- Core performance monitoring - 3D Monicore (Units 2 and 3)
- Digital feedwater control (Unit 3)
- Turbine vibration monitor (Unit 3)

These systems had been documented in NUREG-1706, "Year 2000 Readiness in U.S. Nuclear Power Plants," as requiring additional actions to achieve Y2K readiness. These actions were completed on or before October 27, 1999.

## IV. Plant Support

### R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Controls (Program Changes)

a. Inspection Scope (83750)

The inspectors reviewed selected radiological controls program changes since the previous inspection in this area. Areas reviewed included organization and staffing, facilities and equipment, and procedure changes. The inspectors reviewed work in progress, reviewed applicable documentation, and interviewed cognizant personnel.

b. Observations and Findings

There were no program changes identified that adversely affected the radiation protection program. PECO had recently selected a qualified, permanent replacement to fill the vacant Radiological Engineering Manager position. PECO expected to fill additional positions, in part, to staff for increased responsibilities in the area of industrial safety. A defined radiation protection organization was established for the outage and no adverse performance changes were observed.

c. Conclusions

PECO continued to maintain its radiation protection program. No changes were identified that adversely affected radiation protection program performance.

R1.2 Unit 3 Refueling Outage Radiological Controls (ALARA) Planning and Performance

a. Inspection Scope (83750)

The inspectors selectively reviewed the ALARA planning and preparation efforts for the refueling outage. The inspectors reviewed radiological control records; interviewed licensee representatives relative to outage planning; and observed activities to determine the effectiveness of planning, preparation, and management oversight for radiologically challenging work activities. The inspectors reviewed (either through review of documentation or direct observation) selected work activities that had the potential for creating radiological hazards (e.g, refueling activities, reactor water cleanup valve work, and control rod drive removal). The inspectors also reviewed radiation performance during spent fuel pool diving activities conducted earlier in 1999.

b. Observations and Findings

PECO performed overall effective planning and preparation for outage radiological work activities, met its 1999 outage ALARA goals, and implemented good efforts to reduce personnel occupational exposure for work activities to as low as is reasonably achievable. PECO monitored areas of the station that could exhibit elevated radiation levels during draining of the reactor cavity. Video cameras were extensively used to remotely monitor areas of elevated ambient radiation levels. Work activities for tasks that could result in elevated aggregate personnel radiation exposure were controlled by radiation work permits with detailed ALARA reviews and clearly defined radiation exposure reduction methods. Tours of the station identified overall very good ALARA controls. PECO performed active review of ongoing work to identify anomalies and implement enhanced oversight. PECO identified weaknesses in planning and performance for work on the Unit 3 reactor water clean-up system and placed this matter into its corrective action program.

c. Conclusions

Overall, PECO implemented an effective ALARA program. There were overall effective planning and preparation for outage radiological work activities. PECO met its 1999 outage ALARA goals and implemented good efforts to reduce personnel occupational exposure for work activities to as low as is reasonably achievable. Identified weaknesses were placed in the corrective action program.

R1.3 Unit 3 Refueling Outage Radiological Controls (Internal and External Exposure Controls)

a. Inspection Scope (83750)

The inspectors reviewed records, interviewed cognizant licensee personnel, and observed occupational exposure control practices during licensee work activities and tours of the RCA. The inspectors reviewed high radiation area controls, general radiological posting, implementation of the radiation work permit (RWP) program, and implementation of the dosimetry program. The inspectors toured the drywell, refueling floor, and reactor building and observed ongoing activities and radiological conditions. The inspectors selectively made independent radiation measurements to verify licensee results and reviewed selected work activities that had the potential for creating radiological hazards (e.g, reactor head removal, refueling activities, reactor cavity drain down, reactor water cleanup valve work, and control rod drive removal).

b. Observations and Findings

PECO made appropriate radiological surveys to support planning and preparation of work. Appropriate job-coverage surveys were also made to monitor ongoing work. Calibrated and checked survey instrumentation was used for pre-work and ongoing radiological surveys. Personnel dosimetry was properly issued, worn, and moved to points of highest expected radiation exposure of the body. Multiple dosimetry was used as appropriate. Radiation work permit attachments provided clear guidance for workers and were properly implemented by workers. Selective verification identified that workers were properly signed in on their assigned RWPs. No significant unplanned personnel external or internal exposures were identified.

Engineering controls were effectively used to minimize airborne radioactivity. Selected workers, wearing respiratory protective equipment, were verified to have received training, fit testing, and medical certification to wear the equipment. PECO evaluated low level intakes of radioactive materials using data from air samples and whole body counters, as appropriate. Air sampling was conducted for tasks with the potential to generate airborne radioactivity. No significant airborne radioactivity was identified and no individuals sustained any significant airborne radioactivity intake.

Access points to areas of elevated radiation levels or areas exhibiting contamination were properly posted and barricaded. Appropriate access controls were implemented for High Radiation Areas, including those areas meeting criteria to be locked. PECO implemented effective applied radiological controls for spent fuel pool diving activities.

There was overall effective oversight of work activities by radiation protection personnel and work supervisors. Radiation protection personnel, monitoring reactor cavity drain down, to support re-installation of the reactor vessel head, identified potential elevated dose rate concerns if the reactor cavity was fully drained for the planned work activity. PECO suspended drain down and moved fuel elements in the spent fuel pool away from the spent fuel pool gate to reduce potential dose rates. No elevated radiation dose rates were identified in locations accessible to personnel, including the upper portions of the drywell, which was locked. This matter was included in the licensee’s corrective action program.

c. Conclusions

Applied radiological controls for ongoing work activities were generally well implemented. No significant unplanned personnel external or internal exposures were identified. No significant airborne radioactivity was identified and no individuals sustained any significant airborne radioactivity intake. There was overall effective oversight of work activities by radiation protection personnel and work supervisors. Workers were properly signed in on their assigned RWP.

R1.4 Refueling Outage Radiological Controls (Control of Radioactive Materials and Contamination)

a. Inspection Scope (83750)

The inspectors selectively reviewed radioactive material and contamination control practices, including the adequacy of supply, maintenance, calibration and performance checks of survey and monitoring instruments; the use of personal contamination monitors and friskers; and application of hot particle contamination monitoring. The inspectors also reviewed the current status of non-contaminated systems that interface with contaminated systems.

The evaluation of licensee performance in this area was based on observations during station tours, discussion with cognizant personnel, and review of documentation.

b. Observations and Findings

Radioactive material was properly labeled, stored, and controlled. Contamination monitoring equipment was observed to be operable, within calibration, and properly used by personnel. Radiation and contamination surveys were observed to be comprehensive and detailed. PECO used appropriate contamination control techniques and evaluated detection capability and efficiency of its contamination monitors. PECO used multiple monitor types to monitor personnel exiting the radiological controlled area. Personnel were also monitored for contamination during exit of the Security Building. Personnel identified as contaminated were decontaminated, as appropriate, with appropriate dose evaluations conducted. There were minimal instances of personnel contamination during the outage and the instances were tracked with corrective actions taken to prevent recurrence. There were no significant doses associated with personnel contamination. PECO implemented a new initiative this outage to track buildup of contamination and promptly react to instances of low-level contamination.

PECO performed either monthly or quarterly surveys, based on contamination potential, of areas outside the radiological control areas to identify instances of inadvertent or improper release of radioactive contamination from the radiological control area (e.g., equipment). Dose rate surveys were performed and loose contamination surveys were made of areas such as administration building, security guard house, warehouse, clean shops, and clean tool room. Other than an isolated instance of trace contamination of a ladder noted in early 1998, no significant contamination was identified outside the radiological controlled area. PECO had conducted extensive surveys of tools and equipment within the protected area after the instance with no other examples noted. The instance was entered into PECO’s corrective action program. No instances of uncontrolled radioactive contamination was identified in areas outside the radiological controlled area in 1999.

PECO was also collecting and analyzing soil samples at onsite locations where excavations were occurring near the protected area boundary (e.g., modifications to support fuel cask transport). No contamination was identified in these locations. PECO performed either live-time monitoring or collected and analyzed grab samples of systems which interfaced with contaminated systems to evaluate potential inter-system cross contamination. PECO was conducting sampling and analysis of the systems relative to guidance in NRC Bulletin 80-10, “Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release to Environment.” PECO had recently detected trace radioactive contamination in its Unit 2 reactor building closed-loop cooling water system (RBCCW), initiated an investigation as to its source, and was closely monitoring system contamination. The system is identified in the UFSAR as a system which may experience some cross contamination. The system that cools RBCCW (i.e., service water) was monitored by a live time radiation monitor downstream of the RBCCW heat exchanger. The monitor showed no unusual activity. PECO had conducted a safety evaluation, consistent with NRC Bulletin 80-10 for a previous leak, to support continued operation of the RBCCW system. No other active inter-system contamination was identified, including the Unit 3 RBCCW. PECO had also been continuously sampling (by composite sampling) the circulation water discharge and had not detected any contamination in this discharge which cools the main condensers. The sampling location is located down stream of the discharge point of the mechanical draft cooling towers.

PECO had not experienced any recent unplanned releases or spills of liquid from onsite process building or tanks to the environment. PECO was periodically monitoring storm drain liquids for radioactivity. In early 1999, PECO identified the inadvertent release of a quantity of oil, with trace radioactive contamination, to an offsite oil processor. PECO placed this matter into its corrective action system (PEP No. I0009898), took immediate actions, and concluded the quantity of radioactivity contained in the oil was below Department of Transportation levels requiring specific shipping controls. PECO also performed offsite dose analyses and concluded that no significant offsite dose was received by workers or the public. PECO plans to include the release in its radiological effluent release report.

c. Conclusions

Overall, PECO implemented an effective radioactive material and contamination control program. Radioactive material was properly labeled, stored, and controlled; contamination monitoring equipment was operable, within calibration, and properly used by personnel; and radiation and contamination surveys were observed to be comprehensive and detailed. There were minimal instances of personnel contamination during the outage, the instances were tracked with corrective actions taken to prevent recurrence, and the instances did not result in any significant personnel exposure.

## R7 Quality Assurance in RP&C Activities

a. Inspection Scope (83750)

The inspectors selectively reviewed quality assurance activities including, as appropriate, audits, surveillances, and self-assessment activities.

b. Observations and Findings

PECO implemented an active audit, surveillance, and self-assessment program. PECO was performing ongoing assessments of station radiological controls activities. Most recently, PECO implemented a pilot process to conduct ongoing assessments of station radiological controls activities using a quarterly Master Assessment Schedule. These assessments were being conducted instead of conducting one large assessment every two years. The third quarter assessment of health physics, chemistry and radwaste was of good depth and scope in the areas assessed. The results of the on-going assessments were incorporated into a quarterly continuous assessment report for station management.

The recently completed dosimetry/bioassay, respiratory protection, health physics operations, and ALARA assessment exhibited good scope and depth, and was performance based. Items requiring corrective action were placed in PECO's corrective action program.

The radiation protection organization was conducting individual process self-assessments in specific targeted areas. Self-assessment records were completed which included evaluation methods, identified performance gaps and conclusions/corrective actions.

c. Conclusions

PECO implemented effective assessments of ongoing radiological controls activities. The assessments were of good scope and depth and were performance based.

## R8 Miscellaneous Issues

R8.1 Plant Tour Observations - Radiological Controlled Areas

a. Inspection Scope (83750)

During the inspection period, the inspectors made several tours of radiological controlled areas and observed general housekeeping and station material conditions.

b. Observations and Findings

Overall, housekeeping in the areas toured was very good. Walkways were unobstructed and potentially contaminated materials were clearly marked and segregated. No buildup of combustibles was observed.

c. Conclusions

PECO implemented, overall, very good housekeeping within radiological controlled areas.

**S1 Conduct of Security and Safeguards Activities**

a. Inspection Scope (81700)

Determine whether the conduct of security and safeguards activities met the licensee's commitments in the NRC-approved security plan (the Plan) and NRC regulatory requirements. The security program was inspected during the period of October 25-28, 1999. Areas inspected included: Access Authorization program; alarm stations; communications; and protected area (PA) access control of personnel and packages.

b. Observations and Findings

Access Authorization Program. The Access Authorization (AA) program was reviewed to verify implementation was in accordance with applicable regulatory requirements and Plan commitments. The review included an evaluation of the effectiveness of the AA procedures, as implemented, and an examination of AA records for 12 individuals. Records reviewed included both persons who had been granted and had been denied access. The AA program, as implemented, provided assurance that persons granted unescorted access did not constitute an unreasonable risk to the health and safety of the public. Additionally, access denial records and applicable procedures were reviewed to verify that appropriate actions were taken when individuals were denied access or had their access terminated.

Alarm Stations. Operations of the Central Alarm Station (CAS) and the Secondary Alarm Station (SAS) were reviewed. Both alarm stations were determined to be equipped with appropriate alarms, surveillance and communications capabilities. Interviews with the alarm station operators found them knowledgeable of their duties and responsibilities. Observations and interviews also verified that the alarm stations were continuously manned, independent and diverse so that no single act could remove the plant’s capability for detecting a threat and calling for assistance and the alarm stations did not contain any operational activities that could interfere with the execution of the detection, assessment and response functions.

Communications. Document reviews and discussions with alarm station operators determined that the alarm stations were capable of maintaining continuous intercommunications, continuous communications with each security force member

(SFM) on duty, and alarm station operators were testing communication capabilities with the local law enforcement agencies as committed to in the Plan.

Protected Area (PA) Access Control of Personnel and Hand-Carried Packages. On October 26 and 27, 1999, during peak activity periods, personnel and package search activities were observed at the personnel access portal. Positive controls were determined to be in place to ensure only authorized individuals were granted access to the PA and that all personnel and hand-carried items entering the PA were properly searched.

c. Conclusions

The licensee was conducting its security and safeguards activities in a manner that protected public health and safety and that this portion of the program, as implemented, met the licensee's commitments and NRC requirements.

**S2 Status of Security Facilities and Equipment**

a. Inspection Scope (81700)

Areas inspected were: PA assessment aids; PA detection aids and personnel search equipment.

b. Observations and Findings

Assessment Aids. On October 26 and 27, 1999, the effectiveness of the assessment aids was evaluated by observing the PA perimeter on closed circuit television (CCTV), in the Central Alarm Station (CAS) and the Secondary Alarm Station (SAS), respectively. The evaluation of the assessment aids was accomplished by observing, on CCTV, an SFM performing a perimeter patrol. The assessment aids generally had good picture quality, view and zone overlap. Additionally, to ensure Plan commitments were satisfied, the licensee had procedures in place requiring the implementation of compensatory measures in the event the alarm station operator was unable to properly assess the cause of an alarm.

PA Detection Aids. On October 26 and 27, 1999, while observing the assessment aids, testing was also observed of selected intrusion detection zones in the plant protected area. The appropriate alarm was generated in each zone for each test. During the testing of the alarms, it was noted that a minor modification of the alarm placement would enhance the alarm effectiveness. The licensee implemented prompt actions to make the modifications and completed them prior to the end of this inspection. Through observations and review of the testing documentation associated with the equipment repairs, it was verified that repairs were made in a timely manner and that the equipment was functional and effective, and met the commitments in the Plan.

Personnel and Package Search Equipment. On October 26, 1999, both the routine use and the daily operational testing of the licensee’s personnel and package search equipment were observed. Personnel search equipment was being tested and maintained in accordance with licensee procedures and the Plan and personnel and packages were being properly searched prior to PA access.

Observations and procedural reviews determined that the search equipment performed in accordance with licensee procedures and Plan commitments.

c. Conclusions

The licensee's security facilities and equipment were determined to be well maintained and reliable and were able to meet the licensee's commitments and NRC requirements.

**S3 Security and Safeguards Procedures and Documentation**

a. Inspection Scope (81700)

Areas inspected were: implementing procedures and security event logs.

b. Observations and Findings

Security and Program Procedures. Review of selected security program implementing procedures, associated with personnel search, vehicle search and equipment testing verified that the procedures were consistent with the Plan commitments.

Security Event Logs. The Security Event Logs for the previous twelve months were reviewed. Based on this review, and discussion with security management, it was determined that the licensee appropriately analyzed, tracked, resolved and documented safeguards events.

c. Conclusions

Security and safeguards procedures and documentation were being properly implemented. Event Logs were being properly maintained and effectively used to analyze, track, and resolve safeguards events.

**S4 Security and Safeguards Staff Knowledge and Performance**

a. Inspection Scope (81700)

Area inspected was security staff requisite knowledge.

b. Observations and Findings

Security Force Requisite Knowledge. A number of SFMs in the performance of their routine duties were observed. These observations included alarm station operations, personnel and package searches, and exterior patrol alarm response. Additionally, SFMs were interviewed and based on the responses to questioning, it was determined that the SFMs were knowledgeable of their responsibilities and duties, and could effectively carry out their assignments.

Response Capabilities. Review of documentation of contingency response drills and critiques disclosed that the licensee is exercising this portion of the program. The review also disclosed that the licensee is using lessons learned from the drills to modify and refine the response plan to improve its effectiveness.

c. Conclusions

The SFMs adequately demonstrated that they had the requisite knowledge necessary to effectively implement the duties and responsibilities associated with their position.

**S5 Security and Safeguards Staff Training and Qualifications (T&Q)**

a. Inspection Scope (81700)

Areas inspected were security training and qualifications and training records.

b. Observations and Findings

Security Training and Qualifications. On October 27, 1999, T&Q records of 7 SFMs were reviewed. The results of the review indicated that these personnel were trained in accordance with the approved T&Q plan.

Training Records. Through review of training records, it was determined that the records were properly maintained, accurate and reflected the current qualifications of the SFMs.

c. Conclusions

Security force personnel were being trained in accordance with the requirements of the T&Q Plan. Training documentation was properly maintained and accurate and the training provided by the training staff was effective.

**S6 Security Organization and Administration**

a. Inspection Scope (81700)

Areas inspected were management support and staffing levels.

b. Observations and Findings

Management Support. Review of program implementation since the last program inspection disclosed that adequate support and resources continued to be available to ensure effective program implementation.

Staffing Levels. The total number of trained SFMs immediately available on shift met the requirements specified in the Plan and implementing procedures.

c. Conclusions

The level of management support was adequate to ensure effective implementation of the security program, and was evidenced by the allocation of resources to support programmatic needs.

**S7 Quality Assurance (QA) in Security and Safeguards Activities**

a. Inspection Scope (81700)

Areas inspected were: audits, problem analyses, corrective actions and effectiveness of management controls.

b. Observations and Findings

Audits. Surveillances conducted as part of the 1999 QA Security Program Audit were reviewed. Review of the audit checklists and the surveillances that had been conducted as part of the security audit disclosed that they were comprehensive in scope and depth.

Problem Analyses. A review of data derived from the security department's self-assessment program indicated that potential weaknesses were being properly identified, tracked, and trended.

Corrective Actions. Review of corrective actions implemented by the licensee, in response to the QA audits and self-assessment program, disclosed that all corrective actions had been implemented and were effective.

Effectiveness of Management Controls. The licensee had programs in place for identifying, analyzing and resolving problems. They include the performance of annual QA audits, a departmental self-assessment program and the use of industry data, such as violations of regulatory requirements identified by the NRC at other facilities, as a criterion for self-assessment.

c. Conclusions

The review of the licensee's audit program indicated that the audits were comprehensive in scope and depth, that findings were reported to the appropriate level of management, and that the program was being properly administered. In addition, a review of the documentation applicable to the self-assessment program indicated that the program was being effectively implemented to identify and resolve potential weakness.

## V. Management Meetings

### X1 Exit Meeting Summary

The inspectors presented the results of the inspection to members of licensee management on November 22, 1999. The licensee acknowledged the findings presented.

## INSPECTION PROCEDURES USED

| | |
|---|---|
| IP 37550 | Engineering |
| IP 37551 | Onsite Engineering |
| IP 61726 | Surveillance Observation |
| IP 62707 | Maintenance Observation |
| IP 71707 | Plant Operations |
| IP 73753 | Inservice Inspection |
| IP 81700 | Physical Security Program for Power Reactors |
| IP 83750 | Radiation Protection |
| IP 92903 | Follow-up Engineering |
| TI 2515/141 | Review of Year 2000 (Y2K) Readiness of Computer Systems at Nuclear Power Plants |

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 50-277/99-08-01 | NCV | Unit 2 Reactor Coolant System Heatup Greater Than 100°F/hour |
| 50-278/99-08-02 | NCV | Failure to Adhere to Procedural Requirements in the Performance of Ultrasonic Testing of Safety-Related Components |
| 50-278/99-08-03 | EEI | Failure to Include Two Core Spray System Welds in the ISI Program |

Closed

| | | |
|---|---|---|
| 50-277/99-08-01 | NCV | Unit 2 Reactor Coolant System Heatup Greater Than 100°F/hour |
| 50-278/99-08-02 | NCV | Failure to Adhere to Procedural Requirements in the Performance of Ultrasonic Testing of Safety-Related Components |
| 50-277/98-09-01 | VIO | Failure to Translate Design Basis Information Into Procedures for the Reactor Core Isolation Cooling (RCIC) System |
| 50-278/3-99-004 | LER | Unplanned Engineered Safety Feature Actuations During Planned Modification Activities in the Main Control Room |
| 50-278/3-99-005 | LER | Unplanned Engineered Safety Feature Actuations During Planned Modification Activities in the Main Control Room |

Discussed

| | | |
|---|---|---|
| 50-277/2-99-006 | LER | Engineering Safety Feature Actuation Following the Turbine Trip and the Requirements of 10 CFR 50.73 (A)(2)(i)(B) for Exceeding the Heatup Rate Specified by Technical Specifications |

## LIST OF ACRONYMS USED

| | |
|---|---|
| AA | access authorization |
| ALARA | as low as is reasonably achievable |
| ANII | authorized nuclear inservice inspector |
| AR | action request |
| ASME | American Society of Mechanical Engineers |
| CAS | central alarm system |
| CCTV | closed circuit television |
| DC | direct current |
| EEI | escalated enforcement issue |
| EOP | emergency operation procedure |
| FFD | fitness-for-duty |
| GE | General Electric |
| HCU | hydraulic control unit |
| HP | health physics |
| HPCI | high pressure coolant injection |
| HRA | high radiation area |
| ISI | inservice inspection |
| IVVI | in-vessel visual inspection |
| MT | magnetic particle |
| NCV | non-cited violation |
| NDE | nondestructive examination |
| LOCA | loss of coolant accident |
| LOOP | loss of offsite power |
| NCV | non-cited violation |
| NPSH | net positive suction head |
| NRC | Nuclear Regulatory Commission |
| ODCM | Offsite Dose Calculation Manual |
| PA | protected area |
| PBAPS | Peach Bottom Atomic Power Station |
| PDR | public document room |
| PECO | PECO Energy |
| PEP | performance enhancement program |
| PMT | post-modification testing |
| QA | quality assurance |
| RBCCW | reactor building closed-loop cooling water |
| RCA | radiologically controlled area |
| RCIC | reactor core isolated cooling |
| RHR | residual heat removal |
| RP&C | radiological protection and chemistry |
| RPV | reactor pressure vessel |
| RWP | radiation work permit |
| SAS | secondary alarm system |
| SFM | security force member |
| SUN | shift update notice |
| SRV | steam relief valve |
| T&Q | training and qualification |

| | |
|---|---|
| The Plan | NRC-approved physical security plan |
| TI | temporary instruction |
| TS | technical specifications |
| UFSAR | Updated Final Safety Analysis Report |
| UT | ultrasonic |
| VAC | volts alternating current |