December 18, 2000

Mr. Oliver D. Kingsley
Chief Nuclear Officer
PECO Energy Company
1400 Opus Place
Downers Grove, IL 60515-5701

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION - NRC INSPECTION REPORT 05000277/2000-012, 05000278/2000-012

Dear Mr. Kingsley:

On November 18, 2000, the NRC completed an inspection at the Peach Bottom Atomic Power Station. The enclosed report documents the inspection results which were discussed on November 29, 2000, with Mr. Jay Doering and other members of your staff.

This inspection was an examination of 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, the inspectors identified three issues of very low safety significance (Green). Two issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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 facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

If you have any questions, please contact me at 610-337-5233.

Sincerely,

/RA/

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

Docket Nos.: 05000277, 05000-278,
License Nos.: DPR-44, DPR-56

Enclosure: Inspection Report No. 05000277/2000-012, 05000278/2000-012

cc w/encls:
J. Hagan, Senior Vice President, Nuclear Operations
J. Skolds, Chief Operating Officer
J. Doering, Vice President, Peach Bottom Atomic Power Station
G. Johnston, Plant Manager, Peach Bottom Atomic Power Station
J.A. Benjamin, Vice President - Licensing
J.A. Hutton, Director, Licensing, PECO Energy Company
G. Hunger, Chairman, Nuclear Review Board
R. Boyce, Director, Nuclear Quality Assurance
A. F. Kirby, III, External Operations - Delmarva Power & Light Co.
A. A. Winter, Manager, Experience Assessment
J. W. Durham, Sr., Senior Vice President and General Counsel
H. C. Kresge, Manager, External Operations, Connectiv
N. J. Sproul, Manager, Financial Control & Co-owner Affairs, Connectiv
R. McLean, Power Plant Siting, Nuclear Evaluations
D. Levin, Acting Secretary of Harford County Council
R. Ochs, Maryland Safe Energy Coalition
J. H. Walter, Chief Engineer, Public Service Commission of Maryland
Mr. & Mrs. Dennis Hiebert, Peach Bottom Alliance
Mr. & Mrs. Kip Adams
Commonwealth of Pennsylvania
State of Maryland
TMI - Alert (TMIA)

Distribution w/encls:
Region I Docket Room (with concurrences)
A. McMurtray - NRC Resident Inspector
H. Miller, RA
J. Wiggins, DRA
C. Cowgill, DRP
D. Florek, DRP
D. Cullison, DRP
C. O'Daniell, DRP
D. Holody, ORA
W. Borchardt, OE
J. Shea, OEDO
E. Adensam, NRR
J. Clifford, NRR
J. Boska, NRR

DOCUMENT NAME: C:\PB 2000-012 Rev 2.wpd
After declaring this document "An Official Agency Record" it **will** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP	RI/DRP	/		
NAME	BWelling	CCowgill			
DATE	12/18/00	12/18/00	01/ /01	01/ /01	01/ /01

OFFICIAL RECORD COPY

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket Nos.	05000277 05000278
License Nos.	DPR-44 DPR-56
Report Nos.	05000277/2000-012 05000278/2000-012
Licensee:	PECO Energy Correspondence Control Desk P.O. Box 160 Kennett Square, PA 19348
Facility:	Peach Bottom Atomic Power Station Units 2 and 3
Inspection Period:	October 1, 2000 through November 18, 2000
Inspectors:	B. Welling, Acting Senior Resident Inspector M. Buckley, Resident Inspector J. Jang, Senior Radiation Specialist R. Bhatia, Reactor Inspector
Approved by:	Curtis J. Cowgill, Chief Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

Peach Bottom Nuclear Power Plant
NRC Inspection Report 05000277/2000-012, 05000278/2000-012

IR 05000277/2000-012, 05000278/2000-012; on 10/01-11/18/2000; PECO Energy; Peach Bottom Atomic Power Station; Units 2&3. Operability Evaluations.

The inspection was conducted by resident inspectors and a regional radiation specialist. The inspection identified three Green findings and two Non-Cited Violations. The significance of all findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No color" or by the severity level of the applicable violation.

A. <u>Inspector Identified Findings</u>

Cornerstone: Mitigating Systems

- Green. Emergency service water (ESW) system check valve 2-33-514 failed open, allowing safety-related ESW flow to be partially diverted from emergency diesel generators (EDGs) and emergency core cooling system room coolers. The inspectors and the licensee identified that this risk important component had not been included in a preventive maintenance program.

  This issue caused the ESW system and the EDGs to be degraded for a period of up to two years. This finding was of very low safety significance because, although the ESW flow rate to the EDGs was below the design basis minimum value, engineering personnel determined that the EDGs would have remained available during accident conditions. (Section 1R15)

- Green. Unit 2 operators experienced four unplanned, unexpected isolations of shutdown cooling during refueling outage 2R13. During one time period in which there were three repetitive isolations, the reactor coolant system (RCS) temperature rose from 153 degrees to 171 degrees. The inspectors identified a corrective action performance issue, in that previous isolations of shutdown cooling had not been fully investigated and resolved.

  This finding was of very low safety significance because the increase in RCS temperature did not constitute a loss of control and did not require phase 2 analysis per the guidance in NRC Manual Chapter 0609, Appendix G. In all instances, operators were able to restore the shutdown cooling system promptly. (Section 1R15)

Cornerstone: Barrier Integrity

- Green. Unit 2 'H' torus to drywell vacuum breaker failed open during stroke testing. Operators shut down the unit as required by technical specifications.

The inspectors identified a Non-Cited violation for an inadequate preventive maintenance procedure.

This finding was of very low safety significance because, although the primary containment was rendered inoperable, the vacuum breaker was only partially open for a duration of approximately 19 hours. (Section 1R15)

Cornerstone: Other Activities

B. Licensee Identified Violations

One violation of very low significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. This violation is listed in section 4OA7 of this report.

TABLE OF CONTENTS

SUMMARY OF FINDINGS ii

TABLE OF CONTENTS iv

SUMMARY OF PLANT STATUS 1

1. REACTOR SAFETY 1
1R01 Adverse Weather Protection 1
1R04 Equipment Alignment 2
1R05 Fire Protection 2
1R06 Flood Protection Measures 3
1R11 Licensed Operator Requalification 3
1R12 Maintenance Rule Implementation 3
1R13 Maintenance Risk Assessment and Emergent Work Evaluation 4
1R14 Personnel Performance During Non-Routine Plant Evolutions 4
1R15 Operability Evaluations 5
1R17 Permanent Plant Modifications 8
1R19 Post-Maintenance Testing 8
1R20 Refueling and Outage Activities 9
1R22 Surveillance Testing 9

2. RADIATION SAFETY 10
2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems 10

4. OTHER ACTIVITIES 11
4OA1 Performance Indicator Verification 11
4OA2 Identification and Resolution of Problems 11
4OA3 Event Follow-up 11
4OA5 Other 12
4OA6 Meetings 12
4OA7 Licensee Identified Violation 12

ITEMS OPENED, CLOSED, AND DISCUSSED 13

ATTACHMENT 1 14

Report Details

SUMMARY OF PLANT STATUS

UNIT 2

Unit 2 began this inspection period shutdown in refueling outage 2R13.

October 4	Unit 2 reactor was taken critical.
October 5	Operators halted the reactor startup following the discovery of a missed post-maintenance test on a control rod. Following satisfactory testing, operators re-commenced the startup.
October 10	The unit reached 100% power.
October 17	Operators reduced power to approximately 65% to repair a condenser tube leak. The unit was restored to 100% on October 18.
October 22	Unit load was reduced to 16% due to an inoperable torus/drywell vacuum breaker.
October 23	The unit was shutdown to repair the torus/drywell vacuum breaker. The reactor was taken critical on October 24 and unit load was 100% on October 26.
November 13	Operators reduced load to 79% to repair the 2C circulating water pump traveling screen. The unit was restored to 100% on the same day.

UNIT 3

Unit 3 began this inspection period at approximately 35% power, following a power reduction in response to a low oil level in the 3B recirculation pump motor.

October 1	Unit load was restored to 100%.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee's preparations for cold weather conditions and walked down selected systems that could be affected by low temperature to verify that these systems would remain functional during cold weather conditions. The inspectors used RT-O-040-620-2, Rev 8, "Outbuilding HVAC and Outer Screen Inspection for Winter Operation;" RT-O-040-630-2, Rev 7, "Winterizing Procedure;" AO 29.2, Rev 6, "Discharge Canal to Intake Pond Cross-Tie Gate Operation;" and SO 48.4.A, Rev 4, "Draining Emergency Service Water and High Pressure Service Water Return Lines to Emergency Cooling Tower for Winter Freeze Protection" during this inspection.

b. Issues and Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems or trains to verify that they were properly aligned for operation. The inspection verified critical portions of redundant or backup systems/trains while a system was out of service. The inspectors reviewed valve positions, electrical power availability, and the general condition of major system components.

- Emergency diesel generators E1, E3, E4, while E2 was inoperable
- 3B residual heat removal (RHR) system, while the 3A residual heat removal system was inoperable for maintenance

b. Issues and Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed walkdowns of the following plant areas to assess control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures:

- Emergency Diesel Generator Building (Fire area 43)
- Unit 2 High Pressure Coolant Injection System Room (Fire area 2, Zone 59)
- Unit 2 'A' and 'C' Residual Heat Removal System Rooms
- Unit 3 'A' and 'C' Residual Heat Removal System Rooms

b. Issues and Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspector reviewed the licensee's flooding mitigation plans and the condition of associated equipment. The inspector walked down the emergency core cooling compartments, emergency diesel generator rooms, and the high pressure service water/emergency service water compartments. The inspector reviewed:

- the adequacy of watertight doors;
- the sealing of equipment below postulated flood levels;
- the condition of sump pumps and room flooding alarms.

The inspector also verified that there were no unanalyzed sources of flooding, including the common drain systems and sumps between flood areas.

The inspector verified appropriate procedures were in place to require a plant shutdown and to protect safety-related equipment in the event of flooding caused by high river levels. The inspector also reviewed several risk significant alarm response procedures related to compartment flooding. The inspector reviewed performance enhancement program (PEP) I0004994, "Unusual Event Due to High River Flow Rates and Levels."

b. Issues and Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed a licensed operator requalification training simulator scenario to identify discrepancies and deficiencies in training, and to assess licensed operator performance and the evaluator's critique.

b. Issues and Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors interviewed appropriate facility personnel and reviewed documentation to determine whether the selected systems met maintenance rule requirements with respect to: scoping, risk significance, performance criteria, goals, characterization of failures, and corrective action programs. The following system performance issues were reviewed for Units 2 and 3:

- 3 'C' high pressure service water system check valve

- 2D residual heat removal system leak
- 4 KV buses E-22 and E-42 lockout relays
- emergency service water system check valve 2-33-514 failure
- torus/drywell vacuum breaker failure

b. Issues and Findings

No findings of significance were identified. Findings related to the emergency service water system check valve and the torus/drywell vacuum breaker failures are discussed in Section 1R15.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed PECO's risk evaluation and contingency plans for selected planned and emergent work activities to verify that appropriate risk evaluations were performed and to assess the licensee's management of overall plant risk. The inspectors attended planning meetings and discussed the risk management aspect of the activities with operators, maintenance personnel, system engineers, and work coordinators for the following issues:

- E2 emergency diesel generator (EDG) balancing
- Unit 3 reactor core isolation cooling pump bearing low oil level troubleshooting
- E4 EDG special testing and load run

b. Issues and Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed the performance of operations personnel during the following non-routine evolution:

- Unit 2 power reduction and shutdown for an inoperable torus/drywell vacuum breaker

b. Issues and Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed three operability evaluations to ensure that the required Technical Specification actions were satisfied and the component or system remained available so that no unrecognized increase in risk occurred. The inspectors discussed the evaluations with cognizant engineering and operations personnel. The following evaluations were reviewed:

- Emergency service water (ESW) check valve failure and system impact
- Unit 2 torus/drywell vacuum breaker failure
- Multiple isolations of shutdown cooling

b. Issues and Findings

The inspectors identified three GREEN findings, as described in the following sections.

.1 (Closed) LER 2-00-004: Emergency Diesel Generators in a Degraded Condition Potentially Outside of the Design Basis and (Closed) URI 05000277/2000-010-01 Emergency Service Water System Check Valve Failure

NRC Inspection Report 05000277(278)/2000-010 discussed a failure of ESW check valve 2-33-514. Operators determined that the valve was stuck open, creating a flow diversion path for emergency service water away from safety related loads, such as emergency diesel generators and emergency core cooling system coolers.

The report identified the following two issues for further evaluation.

1. The inspectors and operations personnel noted that, during two periods in which the ESW system was declared inoperable, operators did not address the operability status of the EDGs or associated Technical Specifications action statements for Unit 2, which was in Mode 5 (refueling) at the time.

   The inspectors identified that operators did not follow operations department procedures with regard to assessing the operability of the EDGs. Specifically, operators did not follow Nuclear Operators Manual Section NOM-C-11.1, paragraphs 3.5 and 3.7, which state that Operability Reviews should be performed when an ongoing verification calls the status of operability into question and that an immediate declaration of Inoperable shall be made when reasonable expectation of operability does not exist. The inspectors concluded that this constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. Several days after the event, engineering personnel completed an operability determination for the EDGs and concluded that they were operable during the periods in which the ESW system was inoperable.

2. The inspectors determined that this event required further evaluation in the significance determination process (SDP) to determine the safety significance of the valve failure.

The inspectors determined that the valve failure was more than a minor issue since it had a credible potential impact on safety. Specifically, the event reduced cooling flow to the EDGs below the minimum design basis flow rate, which potentially could prevent the EDGs from performing their design function. The last successful test of the valve was in September 1998, so it was potentially in a degraded or inoperable condition for up to two years. However, the issue was of very low significance based on an engineering evaluation that determined that the EDGs would have remained available under accident conditions during the two-year period. A Phase 3 SDP analysis of the event conducted by a Region I Senior Reactor Analyst concluded that the failure of ESW check valve 2-33-514 was a finding of very low safety significance (GREEN).

Engineering personnel concluded the root cause of the event was a check valve disk nut retaining pin material that was subject to corrosion. They believed that the pin most likely corroded and allowed the nut to back off. Replacement materials were under evaluation.

The engineering analysis discussed above utilized system flow rate test data with the as-found condition of the check valve and the recorded ESW (heat sink) temperatures for the two-year period. This as-found test provided valuable information for assessing the significance of this event. The analysis also demonstrated that the event did not result in a Maintenance Rule functional failure.

The inspectors noted that the check valve, although recognized to be a risk important component, was not covered under a preventive maintenance program. The lack of a preventive maintenance program precluded the identification of a degrading condition prior to the check valve failing. The inspectors concluded that this was a maintenance program performance issue and did not constitute a violation of NRC requirements. The licensee documented this issue in PEP I0011797.

.2 Torus/Drywell Vacuum Breaker Failure

On October 22, 2000, the failure of the Unit 2 ‘H’ torus/drywell vacuum breaker to fully close during surveillance testing rendered primary containment inoperable. Operators performed a plant shutdown as required by technical specifications. Inspections of the vacuum breaker found the valve disk and seat misaligned due to a missing hinge pin retaining clip. The retaining clip was replaced.

The system manager determined that the vacuum breaker failure was a Maintenance Rule Functional Failure. In addition, the system manager considered this failure to be a Maintenance Preventable Functional Failure (MPFF), because it could have been prevented through proper maintenance. The inspectors noted that the licensee identified a similar event in 1991 where the same retaining clip on this vacuum breaker was found missing and was replaced during maintenance.

The inspectors determined that the failure of the vacuum breaker was more than a minor issue, as it had a potential credible impact on safety. Specifically, the vacuum breaker failure degraded the integrity of the primary reactor containment, which would impact the ability of primary containment to perform its design function. The duration of the event was approximately 19 hours. Nevertheless, a Phase 3 SDP analysis performed by a Region I Senior Reactor Analyst concluded that the vacuum breaker failure was of very low safety significance (GREEN).

The inspectors and the licensee identified that the vacuum breaker preventive maintenance procedure M-007-001, “L and J Technologies Vacuum Breaker Maintenance,” Revision 8, did not specify an inspection of the hinge pin assembly. The licensee missed an opportunity to include these inspections when the procedure was revised in July 2000.

The inspectors concluded that maintenance procedure M-007-001, “L and J Technologies Vacuum Breaker Maintenance,” Revision 8, was not appropriate to the circumstances, constituting a violation of 10 CFR 50, Appendix B, Criterion V, “Procedures.” Because of the very low safety significance of the item and because the licensee has included it in their corrective action program (PEP I0011883), this procedure violation is being treated as a Non-Cited Violation. (**NCV 05000277/2000-12-01**)

.3 Multiple Isolations of Shutdown Cooling

During the 2R13 refueling outage, operators experienced four unplanned isolations of shutdown cooling (SDC) due to high pressure isolation switch actuations. The licensee preliminarily concluded that the first isolation on September 24 was spurious, as a calibration check of the pressure switches was satisfactory. Three other isolations occurred on October 2 during the performance of a containment Integrated Leak Rate Test (ILRT). Engineering personnel stated that these events were caused, in part, by an ILRT procedure that did not fully account for the reduced operating margin to the high pressure isolation setpoint. Engineers also noted that they needed to investigate the isolation events in further detail to determine if there was a common cause among all four isolations. The inspectors concluded that the engineering review of the ILRT procedure was less than rigorous, which was an important contributing cause of the multiple isolations.

At the time of the isolations during the ILRT, SDC was the only operable decay heat removal system. During this period, the RHR heat exchanger inlet temperature increased from 153 degrees to 171 degrees. While this was a notable increase in temperature, there was still a large temperature margin before boiling would be a concern. In all instances, operators restored the shutdown cooling system promptly.

The inspectors identified that there were previous occurrences of SDC isolations on Unit 2 that were not fully investigated. For example, on October 2, 1999, a similar SDC isolation occurred, but no cause was identified. The pressure switches were found to be in calibration. No PEP corrective action document was initiated. Further, in April 2000, engineering personnel initiated an action item to troubleshoot the isolations, but no action had been taken prior to the outage. The inspectors brought this issue to the attention of engineering management. Engineers also noted that there were two other not-fully-understood SDC isolations on Unit 2 since 1994. The inspectors concluded that engineering personnel had missed opportunities to investigate previous SDC isolations and this constituted a corrective actions performance issue.

The inspectors determined this issue was more than minor because the isolations impacted the ability of the operators to control reactor temperature and caused the shutdown cooling system to be inoperable. However, the finding was of very low safety significance (GREEN) because the increase in reactor coolant temperature during the ILRT isolations did not constitute a loss of control as defined in Table 1 of Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process. Further, the events did not constitute a finding requiring phase 2 analysis. The events were documented in the corrective action program as PEPs I0011763 and I0011821. The inspectors did not identify a violation of NRC requirements.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the following permanent plant modification to verify that the design bases, licensing bases, and performance capability had not been degraded through modification.

- Unit 2 Average Power Range Monitoring System Modification

b. Issues and Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed supporting data and observed portions of the following post-maintenance testing:

- Calibration/functional check of Average Power Range Monitor (APRM) ‘1' MAT P507.CAL11-2 (Unit 2)
- Main steam line high flow trip logic cards
- E4 emergency diesel generator (EDG) starting air receiver relief valve
- 3A residual heat removal (RHR) system planned maintenance (ST-O-010-301-3)

b. Issues and Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors observed and/or reviewed selected activities and controls during the latter portion of refueling outage 2R13, including:

- containment Integrated Leak Rate Testing (ST-J-07A-600-2, Rev 6)

- electrical power alignment
- residual heat removal system operation
- availability of emergency core cooling systems and makeup water sources
- containment controls and integrity
- core verification following core alterations (M-C-797-020 Rev 6)
- reactor startup, including preparations, control rod withdrawal, and reactor coolant system heatup

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed and observed portions of the following surveillance tests, and compared test data with established acceptance criteria to verify the system demonstrated the capability of performing its intended safety functions and its operational readiness:

- Unit 2 Reactor Core Isolation Cooling System Pump, Valve and Flow and Unit Cooler Functional and In-Service Test (ST-O-013-301-2)
- Integrated Leak Rate Testing Review (ST-J-07A-600-2)
- Unit 3 Reactor Core Isolation Cooling System Pump, Valve, Flow and Unit Cooler Functional and In-Service Test (ST-O-013-301-3)
- E4 Emergency Diesel Generator (EDG) Slow Start and Full Load Test

b. Issues and Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Public Radiation Safety

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems

a. Inspection Scope

The inspector reviewed the following documents and conducted the following activities to evaluate the effectiveness of the licensee's radioactive gaseous and liquid effluent control programs. The requirements of the radioactive effluent controls were specified in the Improved Technical Specifications/Offsite Dose Calculation Manual (ITS/ODCM):

- 1998 and 1999 Radiological Annual Effluent Release Reports and Radiation Dose Assessment Reports;
- ODCM, Revision 12, May 17, 2000, and technical justifications and 10CFR50.59 evaluations, for ODCM changes made;
- ODCM updating process (for Revision 13), including technical justifications;
- analytical results for charcoal cartridge, particulate filter, and noble gas samples;
- implementation of the compensatory sampling and analysis program when the effluent radiation monitoring system (RMS) is out of service;
- Walk-down for determining the availability of radioactive liquid/gaseous effluent RMS and standby gas treatment system and for determining the equipment material condition;
- radioactive liquid and gaseous release permits, including burning of radioactive waste oil required by Section 3.8.C of the ODCM;
- associated effluent control procedures, including analytical laboratory procedures;
- calibration records for laboratory measurements equipment;
- implementation of measurement laboratory quality control program;
- self-assessments;
- QA audit (PAR-98-011) for the radiological effluent control/ODCM implementations;
- comparisons of projected public dose calculations between the licensee's computer code and the NRC's PCDose code:

    (1) liquid pathway;
    (2) noble gas pathway;
    (3) iodine pathway;
    (4) particulate pathway; and
    (5) beta and gamma air doses.

- surveillance testing results for control rooms and standby gas treatment system air cleaning systems, required by ITS 5.5.7, Ventilation Filter Testing Program;
- the response to the NRC Generic Letter 99-02, Laboratory Testing of Nuclear-Grade Activated Charcoal, including changed TS;
- effluent RMS channel calibration, functional test, and flow monitor calibration results for the following monitors:

(1) liquid radwaste effluent line radiation monitor and flow monitor (listed in ODCM Section 4.8.B.3.3; 4.8.B.3.5; and 4.8.B.3.6);
(2) Units 2 and 3 service water RMS listed in ODCM;
(3) Units 2 and 3 high pressure service water RMS listed in ODCM;
(4) Units 2 and 3 reactor building closed cooling water RMS; and
(5) Units 2 and 3 reactor vent stacks and main stack (common) noble gas RMS, and flow monitors (listed in ODCM Section 4.8.C.4.4; 4.8.C.4.6; and 4.8.C.4.7).

b. Issues and Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed the accuracy and completeness of the supporting data for the following licensee Performance Indicators. The inspectors reviewed applicable portions operator logs, surveillance testing information, and maintenance records for the period January to September 2000.

- Unplanned Power Changes (Initiating Events Cornerstone)
- High Pressure Injection System Unavailability (Mitigating Systems Cornerstone)

b. Issues and Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

The inspectors identified a corrective actions performance issue associated with the multiple isolations of shutdown cooling, as discussed in Section 1R15 of this report.

4OA3 Event Follow-up

.1 (Closed) LER 2-00-005: Condition Prohibited by Technical Specifications by Entering Mode 2 Without Performing a Required Surveillance Test - Unit 2

This issue is discussed in Section 4OA7 of this report. The inspectors performed an onsite review of this LER.

.2 (Closed) LER 2-00-004: Emergency Diesel Generators in a Degraded Condition Potentially Outside of the Design Basis

This issue is discussed in Section 1R15 of this report. The inspectors performed an onsite review of this LER.

.3 (Closed) LER 2-00-003: Unplanned Manual Reactor Protection System (RPS) Scram

On September 15, 2000, with Unit 2 at approximately 16% power and 24% flow, operators performed a manual reactor scram to prevent operation in the restricted zone of the power to flow map, after an unplanned trip of the 2B reactor recirculation pump. The recirculation pump trip was caused by improper tag-out coordination. The licensee entered this occurrence in their corrective action program as PEP I0011713. The on-site review of this LER identified no findings of significance.

4OA5 Other

Review of Institute of Nuclear Power Operations (INPO) Evaluation Report

The inspectors reviewed the final report of an INPO Evaluation conducted in February 2000. The inspectors identified no new findings of significance.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the results of the inspection to Mr. J. Doering and members of PECO's management on November 29, 2000. PECO management acknowledged the findings presented. No proprietary information was identified.

4OA7 Licensee Identified Violation

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy for being dispositioned as a Non-Cited Violation (NCV):

NCV Tracking Number	Requirement Licensee Failed to Meet
05000277/2000-12-02	Technical Specification SR 3.0.4 states that entry into a mode or other specified condition in the applicability of a limiting condition for operation (LCO) shall not be made unless the LCO's surveillances have been met with their specified frequency. A required surveillance (SR 3.1.4.3, Scram Timing) was not performed on Unit 2 for control rod 34-55 prior to entry into Mode 2. This violation was caused by inadequate tracking of post-maintenance testing. The issue is documented in the licensee's corrective action program as PEP I0011836.

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened/Closed

05000277/2000-12-01	NCV	Inadequate Maintenance Procedure for Torus/Drywell Vacuum Breakers - Unit 2
05000277/2000-12-02	NCV	Condition Prohibited by Technical Specifications by Entering Mode 2 Without Performing a Required Surveillance Test - Unit 2

Closed

05000277/2000-10-01	URI	Emergency Service Water System Check Valve Failure
2-00-003	LER	Unplanned Manual Reactor Protection System (RPS) Scram
2-00-004	LER	Emergency Diesel Generators in a Degraded Condition Potentially Outside of the Design Basis
2-00-005	LER	Condition Prohibited by Technical Specifications by Entering Mode 2 Without Performing a Required Surveillance Test - Unit 2

PARTIAL LIST OF PERSONS CONTACTED

PECO Energy Company

J. Doering, Site Vice President
G. Johnston, Plant Manager
P. Davison, Engineering Director
J. Anthony, Maintenance Director
J. Bouck, Senior Manager, Operations
C. Mudrick, Senior Manager, Plant Engineering
D. Warfel, Senior Manager, Design Engineering
A. Winter, Manager, Experience Assessment
H. Trimble, Radiation Protection Manager

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.