September 6, 2001

Mr. Oliver D. Kingsley
President and CNO
Exelon Nuclear
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION - NRC INSPECTION REPORT 50-277/01-06, 05-278/01-06

Dear Mr. Kingsley:

On August 18, 2001, the NRC completed an inspection at the Peach Bottom Atomic Power Station. The enclosed report documents the inspection findings which were discussed on August 23, 2001, 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 one issue of very low safety significance (Green). This issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny any non-cited violation noted in this report, 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 610-337-5209.

Sincerely,

/RA/

Mohamed Shanbaky, Chief
Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report No. 50-277/01-06 and 50-278/01-06

Attachments: (1) Supplemental Information

cc w/encl:
J. Hagan, Senior Vice President, Mid-Atlantic Regional Operating Group
J. Cotton, Senior Vice President, Operations Support
W. Bohlke, Senior Vice President, Nuclear Services
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 and Regulatory Affairs
M. Gallagher, Director, Licensing, Mid-Atlantic Regional Operating Group
G. Hunger, Chairman, Nuclear Review Board
P. Chabot, Director, Nuclear Oversight
A. F. Kirby, III, External Operations - Delmarva Power & Light Co.
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R. McLean, Power Plant Siting, Nuclear Evaluations
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J. H. Walter, Chief Engineer, Public Service Commission of Maryland
Mr. & Mrs. Dennis Hiebert, Peach Bottom Alliance
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U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket Nos:	50-277, 50-278
License Nos:	DPR-44, DPR-56
Report Nos:	50-277/01-06, 50-278/01-06
Licensee:	Exelon Generation Company, LLC Correspondence Control Desk 200 Exelon Way, KSA 1-N-1 Kennett Square, PA 19348
Facility:	Peach Bottom Atomic Power Station Units 2 and 3
Inspection Period:	July 1, 2001-August 18, 2001
Inspectors:	A. McMurtray, Senior Resident Inspector M. Buckley, Resident Inspector K. Kolaczyk, Engineering Inspector R. Nimitz, Senior Health Physicist L. Cheung, Senior Reactor Inspector L. Prividy, Senior Reactor Inspector P. Kaufman, Senior Reactor Inspector
Approved by:	Mohamed M. Shanbaky, Chief Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000277-01-06, IR 05000278-01-06, on 07/01-08/18/2001; Exelon Generation Company; Peach Bottom Atomic Power Station; Units 2&3. Post-Maintenance Testing

The inspection was conducted by resident inspectors, an engineering inspector, senior reactor inspectors, and a senior health physicist. The inspection identified one Green finding which was considered a non-cited violation. The significance of most findings is indicated by the color (Green, White, Yellow, Red) using Inspection Manual Chapter 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. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- **Green.** A Non-cited violation of 10 CFR 50.55a(f)(4)(ii) and Technical Specification 5.5.6, "Inservice Testing Program" was identified for failure to test the Unit 2 and Unit 3 high pressure coolant injection (HPCI) torus suction check valves for seat leakage in the reverse flow direction. Excessive leakage of these check valves could render the HPCI system inoperable during certain small-break loss of coolant accident scenarios.

  This issue was determined to be of very low safety significance since the respective high pressure coolant injection system remained operable and no actual loss of function occurred. (Section 1R13)

B. Licensee Identified Violations

- Violations of very low significance, which were identified by Exelon, have been reviewed by the inspectors. Corrective actions, taken or planned by Exelon, appeared reasonable. These violations are described in Section 4OA7 of this report.

Report Details

SUMMARY OF PLANT STATUS

UNIT 2

On July 1, 2001, a reactor automatic shutdown occurred from approximately 100 percent power when the electro-hydraulic control (EHC) system power supply failed. Following troubleshooting and repairs, the unit was restarted on July 3 and reached 100 percent power on July 4. Unit 2 operated at approximately 100 percent power for the remainder of the inspection period except for scheduled power changes to support maintenance activities.

UNIT 3

Unit 3 began this inspection period at approximately 98 percent power, in end-of-cycle coastdown, with the fourth and fifth stage feedwater heaters removed from service. Unit 3 ended the inspection period at approximately 81 percent power.

1. REACTOR SAFETY
Initiating Events / Mitigating Systems / Barrier Integrity [Reactor-R]

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed and assessed 10 CFR 50.59 Safety Evaluations (SEs) which were issued prior to the implementation of the new 10 CFR 50.59 program. The reviews were performed to verify that changes made to the facility or procedures as described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59, and that the safety issues pertinent to the changes were properly resolved or adequately addressed. The SEs reviewed covered activities associated with the three cornerstones: initiating events, mitigating systems, and barrier integrity. The inspectors also reviewed screened-out safety evaluations to verify that the screen-out process was appropriately implemented. The specific SEs reviewed are listed in Attachment 1.

The inspectors used the following administrative procedures to understand the method used by Exelon to control the preparation and issuance of the SEs:

- LR-C-13, “10 CFR 50.59 Reviews”, Revision 9, dated March 13, 2001
- LR-CG-13, “Performing 10 CFR 50.59 Reviews,” Revision 4, dated March 13, 2001
- LR-CG-13-2, “10 CFR 50.59 Review Determination Checklist,” Revision 4, dated March 13, 2001
- LR-CG-13-3, “10 CFR 50.59 Screening,” Revision 6, dated March 13, 2001

The inspectors also interviewed engineering personnel engaged in the preparation and the review of the selected 10 CFR 50.59 safety evaluations.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed a partial system walkdown to verify system and component alignment and note any discrepancies that would impact system operability. The inspectors verified selected portions of redundant or backup systems/trains were available while a system was out of service. The inspectors reviewed selected valve positions, electrical power availability, and the general condition of major system components. The walkdown included the following system:

- Unit 2 high pressure coolant injection system during reactor core isolation cooling maintenance outage

b. Findings

No findings of significance were identified.

.2 Complete System Walkdowns

a. Inspection Scope

The inspectors also performed a complete system walkdown to verify that the selected system was properly aligned for operation. The inspectors reviewed valve positions, electrical power availability, and the general condition of major system components. In addition, the inspectors reviewed the Final Safety Analysis Report (FSAR), system design drawings, and issues tracked by the system health report (condition reports, work orders, action requests, and maintenance rule issues). These reviews were conducted to identify discrepancies that could impact system operability. The complete system walkdown was performed on the following:

- Unit 2 residual heat removal system

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors reviewed the Fire Protection Plan and Technical Requirements Manuals to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for the areas examined during this inspection. The inspectors then performed walkdowns of these areas to assess control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The areas included:

- Unit 2 “A” and “C” residual heat removal cubicles
- Unit 2 “B” and “D” residual heat removal cubicles
- Unit 2 135’ Reactor Building
- Unit 3 135' Reactor Building
- Unit 2 High Pressure and Emergency Service Water Bay
- Unit 3 High Pressure and Emergency Service Water Bay

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed the following testing and verified that the test results were adequate to ensure proper heat transfer for the heat exchangers that were tested:

- “Flow Test of Emergency Service Water to the Emergency Core Cooling System Coolers and Emergency Diesel Generator Coolers” (RT-O-033-600-2, Rev. 8)

The inspectors reviewed heat exchanger test methodology, frequency of testing, test conditions, acceptance criteria and trending of results. The inspectors assessed the trending of the measured data for the components inspected and discussed with system managers and technical specialists the proposed actions for any results that were identified not to be within specified acceptance criteria.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

During the weeks of July 16 and July 23, 2001, the inspectors observed the licensed operator performance of three different crews during Licensed Operator Requalification Training Cycle 00-09. The inspectors also observed the evaluator's training critiques of the operators' performance to verify that any operator performance errors were detected and corrected. The inspectors focused on the operating crews' satisfactory completion of crew critical tasks. Critical tasks are limits placed on key reactor plant parameters that will ensure that safety margins are maintained during the simulated malfunctions. Also, the evaluation included the operators' adherence to Technical Specifications, emergency plan implementation, and the use of emergency operating procedures.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the follow-up actions for issues identified on two systems, structures, or components (SSCs) and the performance of these SSCs, to assess the effectiveness of Exelon's maintenance activities. The inspectors verified that problem identification and resolution of these issues had been appropriately monitored, evaluated, and dispositioned in accordance with Exelon's procedures and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, and corrective actions to verify that the actions were reasonable and appropriate. The following issues and documents were reviewed:

Systems

- Units 2 and 3 emergency service water pump room flood level switches and pump structure sump pumps
- Units 2 and 3 reactor vessel instrumentation

Procedures and Documents

- Peach Bottom Maintenance Rule Bases Documentation
- System Health Overview Reports
- Action Request (A1291228)
- AG-CG-028.1, Rev 8, "Maintenance Rule Implementation Program"
- AG-CG-028.1-5, Rev 1, "PECO Energy Approach to Use Maintenance Preventable Functional Failures for Maintenance Rule Performance Monitoring"
- AG-CG-028.1-9, Rev 6, "Guidance for Identifying and Evaluating Maintenance Preventable Functional Failures"

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed risk evaluations and contingency plans for selected planned and emergent work activities to verify that appropriate risk evaluations were performed and to assess Exelon's management of overall plant risk. The inspectors compared the risk assessments and risk management actions against the requirements of 10 CFR 50.65(a)(4) and the recommendations of NUMARC 93-01 Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities." The inspectors verified that risk assessments were performed when required and appropriate risk management actions were identified.

The inspectors attended planning meetings and discussed the risk management of the activities with operators, maintenance personnel, system engineers, and work coordinators to verify that risk management action thresholds were identified correctly. The inspectors also verified that appropriate implementation of risk management actions were performed in accordance with the following Exelon procedures:

- OS-CG-102, Rev 1, "Risk Assessments Using ORAM-Sentinel and Contingency Plan Development"
- AG-CG-043-2, Rev 0, "Peach Bottom Atomic Power Station a(4) System Scope"
- AG-CG-026.9, Rev 3, "Monitoring Performance of Maintenance Activities"
- NOM-C-8.10, Rev 0, "Robust Operational Barriers"

In addition, the inspectors reviewed the assessed risk configurations against the actual plant conditions and any in-progress evolutions or external events to verify that the assessments were accurate, complete, and appropriate for the issues. The inspectors performed control room and field walkdowns to verify that compensatory measures identified by the risk assessments were appropriately performed. The specific plant configurations included:

- Unit 3 high pressure coolant injection (HPCI) suction from torus check valve (CHK-3-23B-61) leakage, including leakage rate testing per Troubleshooting, Rework, and Testing Control Processes (TRTs) No. 01-123 and 01-124
- Loss of numerous station components during an electrical storm on August 10, 2001
- Unit 2 'A' recirculation pump discharge valve (MO-2-02-053A) failure to open

b. Findings

A Non-cited violation of very low safety significance (Green) of 10 CFR 50.55a(f)(4)(ii) and Technical Specification 5.5.6, “Inservice Testing (IST) Program” was identified for failure to test the Unit 2 and Unit 3 HPCI torus suction check valves for seat leakage in the reverse flow direction.

The inspectors determined that Exelon did test both torus suction check valves for full flow capability in the forward flow direction but did not test these valves for seat leakage in the reverse flow direction. Seat leakage in the reverse flow direction is important when HPCI is aligned to pump water from the torus. In this alignment, excessive seat leakage could cause voiding in the HPCI discharge piping. If the HPCI system initiates when the HPCI discharge piping is not full, a water hammer in the discharge piping could prevent HPCI from performing its function. Normally, HPCI is aligned to pump water from the condensate storage tank, the torus supply is isolated, and head pressure from the condensate storage tank keeps the HPCI suction and discharge piping full. Nevertheless, HPCI can be manually aligned to pump water from the torus by the operator and will automatically align to pump water from the torus when the condensate storage tank level is low such as will occur during some design basis accident situations. Site engineering personnel determined that the seat leakage through the Unit 2 and Unit 3 torus suction check valves was not enough to cause a water hammer event that would render the respective HPCI system inoperable.

The failure to test the Units 2 and 3 HPCI torus suction check valves for seat leakage in the reverse flow direction was more than minor because it had a credible impact on safety. Significant leakage in the reverse flow direction could prevent HPCI from performing its function when HPCI is aligned to pump water from the torus. The failure to leak test these valves affected the Mitigating Systems cornerstone since HPCI performs an accident mitigation function. This issue was determined to be of very low safety significance (Green), using the Significance Determination Process (SDP), Reactor Inspection Findings for At-Power Situations, since the respective HPCI systems remained operable and no actual loss of function occurred.

10 CFR 50.55a(f)(4)(ii) and Technical Specification 5.5.6, ”IST Program” requires, in part, that IST of certain ASME Code Class valves shall be performed in accordance with the ASME Boiler and Pressure Vessel Code, OM (the Code). During the current IST Program Test Interval (Third Ten Year Interval), the IST program rules and requirements are in accordance with the 1990 Edition of the ASME OM Code. The 1990 Edition of the Code, Subsection ISTC, Section 1.1 requires IST of valves that support mitigating the consequences of an accident. Contrary to 10 CFR 50.55a(f)(4) (ii) and Technical Specification 5.5.6, IST of the Unit 2 and Unit 3 HPCI torus suction valves, CHK-2-23B-61 and CHK-3-23B-61, was not performed to determine seat leakage in the reverse flow direction. This testing would support maintaining the HPCI system available to mitigate the consequences of an accident when HPCI was aligned to pump water from the torus. This violation of 10 CFR 50.55a(f)(4) (ii) and Technical Specification 5.5.6 is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy. Exelon entered the issue into the corrective

action system as Condition Report (CR) # 00061213. **(NCV 50-277/01-06-01; 50-278/01-06-01)**

1R14 Personnel Performance During Non-Routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed plant computer and recorder data, operator logs and approved procedures and observed control room operators while evaluating the performance of operations personnel in response to the following non-routine evolutions:

- Unit 2 reactor scram due to failure of the electro-hydraulic control system power supply
- Response to an electrical storm on August 10, 2001
- Restoration from the switch of Unit 3 high pressure coolant injection suction from the condensate storage tank to the torus on July 25, 2001

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed three operability evaluations to assess the adequacy of the evaluations, the use and control of compensatory measures, compliance with the Technical Specifications, and the risk significance of the issues. The inspectors verified that the operability determinations were performed in accordance with NOM-C-11.1, Rev. 1, “Operability” and A-C-901, Rev. 10, “Control of Nonconformances.” The inspectors used the Technical Specifications, Technical Requirements Manuals, the Final Safety Analysis Report, and associated Design Basis Documents as references during these reviews. The issues reviewed included:

- Failure of 2A Recirculation Pump Discharge Valve (MO-2-02-053A) to open
- Unit 3 high pressure coolant injection system operability with the torus suction check valve (CHK-3-23B-61) leaking
- E4 emergency diesel generator building main ventilation fan operability while the fan is spinning backwards during operation of the E4 emergency diesel generator supplemental fan

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed 13 selected risk-significant plant modification packages to verify that (1) the design bases, licensing bases, and performance capability of risk significant Structures Systems or Components (SSCs) had not been degraded through modifications, and (2) modifications performed during increased risk configurations did not place the plant in an unsafe condition. The following modification packages were reviewed:

ECR 00-01874	Upgrade of Unit 3 Service Water Valves, December 26, 2000.
ECR 98-02758	P00507 Reactor Stability - Power Range Modification - Unit 3, October, 27, 1998.
ECR 99-02624	E13 4KV Bus Undervoltage Relay Replacement- 3R13, November 24, 1999.
ECR 98-01304	E23 4KV Bus Undervoltage Relay Replacement- 3R12, May 27, 1998.
ECR 99-02547	ECR Required to Put Diodes Across MSIV Solenoids U/3, November 12, 1999.
ECR 99-02131	RWCU Inboard Isolation Valve (MO-3-12-015) - Wiring Changes to Operate the Valve Under Limit Control, September 28, 1999.
ECR 99-01506	Replace MO 3-12-015 with Gate Valve, June 25, 1999.
ECR 00-01726	Add EDG Keep Warm System Isolation Valve, November 22, 2000.
ECR 98-00836	SRV Discharge 3/4" VRV, 10 CFR 50.59 and Documentation Changes, April 8, 1998.
ECR 98-02864	Unit 3 SRV Discharge 3/4" VRV Removal-3R12, November 9, 1998.
ECR 99-00095	Modification to HPSW/ESW Ventilation, January 11, 1999.
ECR 99-00215	Change Unit 3 HWC System Controls to Correspond to 0 to 20 SCFM, January 27, 1999.
ECR 00-00763	Unit 3 Outboard MSIV Steam Leak Based on Sound When Door's Open, May 9, 2000.

Of the 13 plant modifications reviewed, one modification was in the initiating events cornerstone, seven were in the mitigating systems cornerstone, and five were in the barrier integrity cornerstone.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors observed portions of post-maintenance testing activities and reviewed selected test data. The inspectors assessed the adequacy of the test methodology based on the scope of maintenance work performed and the acceptance criteria to demonstrate that the tested components satisfied the design and licensing bases and Technical Specification requirements. The specific tests reviewed included:

- Historical review of recirculation pump discharge motor-operated isolation valve (MO-2-02-53A) testing after planned actuator maintenance during refueling outage, 2R13 (ST-O-094-400-2, Rev 1, "Stroke Time Testing of Valves for Post-Maintenance Testing")
- Unit 2 reactor core isolation cooling (RCIC) system testing after planned maintenance (ST-O-013-301-2, Rev 20, "RCIC Pump, Valve, Flow and Unit Cooler Functional and In-Service Test")

b. 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 systems demonstrated the capability of performing their intended safety functions. The inspectors also verified that the systems and components maintained their operational readiness, met applicable Technical Specification requirements, and were capable of performing their design basis functions. The observed or reviewed surveillance tests included:

- Unit 3 'A' Residual Heat Removal Loop Pump, Valve, Flow, and Unit Cooler Functional and Inservice Test (ST-O-010-301-3, Rev 19)
- Unit 3 'B' Residual Heat Removal Loop Pump, Valve, Flow, and Unit Cooler Functional and Inservice Test (ST-O-010-306-3, Rev 20)
- Unit 3 Local Leak Rate Testing (LLRT) - Panel 30S199 Electrical Penetrations (ST/LLRT 30.07A.12, Rev 4)
- Unit 3 Reactor Core Isolation Cooling (RCIC) Pump, Valve, Flow and Unit Cooler Functional and In-Service Test (ST-O-013-301-3, Rev 19)

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Occupational Radiation Safety [OS]

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

The inspectors conducted the following activities and reviewed the following documents to determine the effectiveness of access controls to radiologically significant areas:

- An inventory was conducted of keys to access points to High and Very High Radiation Areas to determine if: 1) the keys were controlled in accordance with administrative controls, 2) the controls were adequate to prevent unauthorized access, and 3) the keys were present, or signed out, as appropriate.

- Ten access points to locked High Radiation Area were reviewed and challenged to determine if controls were sufficient to preclude unauthorized entry.

- The ambient radiological source term was reviewed to determine if any significant changes in the radiological source term were identified and evaluated.

- In-field portable radiation survey instrumentation was selectively reviewed to determine if the instruments were calibrated and checked before use and that records were up-to-date.

- A review of license provisions that would allow the licensee to provide radiation instrumentation calibration services to a non-10 CFR Part 50 licensee.

- The adequacy of radiological controls for loading of a TN-68 spent fuel storage cask during the week of July 16, 2001, was reviewed and, in addition, a review was conducted of turbine shielding activities.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspectors selectively reviewed the adequacy and effectiveness of the program to reduce occupational radiation exposure to as low as is reasonably achievable (ALARA). The inspector conducted the following activities and reviewed the following documents to determine the effectiveness of ALARA planning and controls:

- The licensee's evaluation of the cause of the unexpected elevated radiation levels encountered in the Unit 2 reactor drywell in September 2000 was reviewed including the licensee's contingency planning in the event a similar increase is encountered in the Unit 3 reactor drywell during the upcoming outage.

- Five planned work activities, likely to result in the highest personnel collective exposures during the upcoming Unit 3 refueling outage, were reviewed to evaluate the adequacy of ALARA planning for the activities. Planned activities reviewed included scaffolding installation, shielding activities, jet pump cleaning, refueling floor activities, and major valve work.

- Plant collective exposure history, current exposure trends, ongoing and planned activities, and the station's two year and three year rolling average collective dose data were reviewed to assess current performance and exposure control challenges.

- The site specific historical trends and current status of tracked source terms were reviewed to determine if the overall plant source term was increasing, stable or declining, and to identify licensee dose rate reduction priorities and reduction strategies.

- ALARA goals, dose reduction initiatives, and the current initiatives to reduce occupational exposure, were reviewed to evaluate efforts in these areas.

- Recent items included in the licensee's self-assessment and corrective action program were reviewed including PEPs IOO12764, IOO12340, IOO12320, continuous assessment report NOSA-PB-01-1Q, and ongoing continuous assessment NOSA-PB-01-2Q.

- The following documents were reviewed to evaluate the licensee's radiation dose reduction initiatives:
  - Contact radiation dose rates for primary piping located within the drywell (Outages 3R08 thru 3R12)
  - Peach Bottom Unit 2 Drywell Dose Rate Increase Root Cause Analysis, dated December 1, 2000
  - Peach Bottom Refueling Outage Reports
    - 3R12, dated October 1999
    - 2R13, dated October 2000
  - Five Year Rolling (2001-2005) Exposure Reduction Plan
  - Various Executive and Station ALARA Committee Meeting minutes

b. <u>Findings</u>

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed the station's records to assess the accuracy and completeness of selected NRC performance indicator (PI) data. The records reviewed included selected Technical Specification limiting condition for operation logs, system surveillance tests, licensee event report, and condition reports. The specific indicators included:

- Unit 2 Residual Heat Removal Safety System Unavailability
- Unit 3 Residual Heat Removal Safety System Unavailability

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed problem reports associated with 10 CFR 50.59 issues and plant modification issues to ensure that the licensee was identifying, evaluating, and correcting problems associated with these areas and that the corrective actions for the issues were appropriate. The inspectors used the following Exelon administrative procedures related to problem identification and resolution:

- LR-C-10, "Performance Enhancement Program (PEP)" Revision 11, dated April 3, 2000
- LR-CG-10, "Performance Enhancement Program" (used to implement the PEP), Revision 4, dated April 3, 2000.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 Emergency Diesel Generator Cross Flow Issue

a. Inspection Scope

The inspectors reviewed Exelon's actions in response to Unresolved Item (URI) 50277;278/00-008-01 and Licensee Event Report (LER)2-00-002. Both items pertained to cross-flows between EDG coolers.

b. Findings

The URI was identified during an August 2000 heat sink inspection (NRC Inspection Report 05000277(278)/2000-008). It pertained to a problem identified by Exelon that under certain high-river-water-temperature conditions, some of the EDGs might not be able to be fully loaded. The problem was caused by undesirable cross-flows of coolant between the jacket water cooler and the intake air cooler. The immediate corrective action taken by Exelon was to close the existing isolation valve (one for each EDG) between the two coolers to reduce the cross-flow. During early spring 2000, Exelon installed an additional normally-closed isolation valve (one for each EDG) under Modification 00-01726 to further reduce the cross flows. Maintaining these valves closed would restore the capability of the EDGs to meet the design basis requirements at a river water temperature up to 90°F (design basis requirement). During the inspection, the inspectors observed these isolation valves closed on all four EDGs.

On August 31, 2000, the licensee issued LER 2-00-002, stating that three of the four EDGs were inoperable during the summer of 1999, based on their inability to mitigate a postulated loss-of-coolant-accident plus loss-of-off-site-power design basis accident for a maximum of approximately 25 continuous hours. The licensee attributed the cause of the event to be an original design deficiency on the EDGs, which allowed cross-flows between the jacket water coolers and the intake-air coolers.

This issue, three of four EDGs inoperable for 25 continuous hours, was determined to be a more than minor issue because multiple inoperable diesels would have a credible impact on safety. This issue affects both the Mitigating System and Barrier cornerstones because the issue affected the operability of a system in a mitigating system as well as equipment necessary for containment heat removal. This issue was assessed using the Significance Determination Process (SDP), Reactor Inspection Findings for At-Power Situations. The phase 1 screening determined that a phase 2 risk evaluation was required because the diesel generators provide emergency power for equipment in both the mitigating system and barrier cornerstones. Using the Peach Bottom reactor risk-informed notebook, the inspectors determined that this issue was of very low safety significance (Green). The reason that this issue is of very low safety significance is because of the low frequency of a loss of offsite power event coupled with and the loss of one or more EDGs, the availability of the Conowingo dam station power source to operate plant safety equipment in the event the EDGs were lost, and the short duration of time during the year when the service water temperature is sufficiently high to adversely affect EDG operation. The NRC inspectors also reviewed a risk analysis performed by Exelon's PRA staff, using the Peach Bottom full scale PRA model, which also confirmed that the safety significance of this issue was very low.

A licensee-identified violation associated with this issue is discussed in Section OA7 of this report. The licensee had issued PEP report I0011529, entitled "EDG potentially outside design basis with heat sink temperature > 80°F", to document the corrective actions associated with this issue.

URI 5000277(278)/2000-008-001 and LER 50-277(278)/2-00-002 are closed.

.2 Loss of Offsite Power Source

a. Inspection Scope

The inspectors performed an onsite review of License Event Report (LER) 2-01-001: Loss of Offsite Power Source Results in Specified System Actuation and Safety System Functional Failure.

b. Findings

This issue is also discussed in Section 4OA7 of this report. The issue was of very low safety significance based on: the two emergency diesel generators (EDGs) being misaligned for only approximately three hours; the EDGs were able to be aligned manually; and the other two EDGs were always fully available for automatic and manual loading during the event. A Phase 2 analysis using the Reactor Inspection Findings for At-Power Situations Significance Determination Process (SDP) concluded that the procedural deficiency that resulted in the loss of automatic loading capability of the two EDGs to two emergency buses on each unit was of very low safety significance. The violation discussed in Section 4OA7 was due to an inadequate station procedure for responding to a loss of an offsite power source. All station procedures identified with inadequate instructions for manipulating emergency buses in response to a loss of an offsite power source were changed.

LER 50-277(278)/2-01-001 is closed.

.3 Unit 2 Automatic Reactor Shutdown

a. Inspection Scope

On July 1, 2001, Unit 2 automatically shutdown from 100% power when an electro-hydraulic control system malfunction caused the main turbine control valves to partially close initiating a turbine trip and a reactor scram. This malfunction was the result of a degrading power supply which caused variations in the servo currents and actual movement of the main turbine control valves. Unit 2 also experienced Groups II and III primary containment isolation valve closures due to decreased reactor water level as a result of the reactor scram.

The inspectors observed plant parameters and status following the automatic reactor shutdown and reviewed strip charts for key reactor parameters. The inspectors also reviewed Check-Off List (COL) GP-18, Revision 32, “Scram Review Procedure Check List” and discussed the automatic reactor shutdown with several operation and engineering managers and staff. The inspectors verified that no significant anomalies of plant parameters occurred during or following the shutdown.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the results of the inspection to Mr. J. Doering and members of Exelon's management on August 23, 2001. Exelon management acknowledged the findings presented. No proprietary information was identified.

4OA7 Licensee Identified Non-compliance

The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as Non-Cited Violations (NCV).

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-277/01-06-02 **NCV 50-278/01-06-02**	Peach Bottom Technical Specifications (TS) Section 3.8.1 requires all EDGs to be capable of supplying onsite Class 1E electrical power, and TS Section 3.8.1.F requires all but one EDG to be restored to operable status within two hours if two or more EDGs are inoperable. During the summer of 1999, three of the four EDGs were inoperable due to cross-flows between the jacket water coolers and the intake air coolers for a maximum of approximately 25 continuous hours. The corrective actions for this violation were already in the licensee's corrective action program (PEP report I0011529). This is being treated as a Non-Cited Violation.
NCV 50-277/01-06-03 **NCV 50-278/01-06-03**	Technical Specification 5.4.1 requires written procedures be established, implemented, and maintained covering activities listed in Regulatory Guide 1.33. Regulatory Guide 1.33 includes abnormal conditions such as loss of electrical power sources. In June 2001, the procedure, SO 53.7.D, "Response to a Loss of #343 Off-Site Startup Source," Revision 24 did not direct proper alignment of emergency bus breaker switches as required to maintain automatic emergency diesel generator power to all emergency buses. Therefore, equipment powered by these buses would not fulfil their safety function to mitigate the consequences of an accident. The corrective actions for this violation were already in the licensee's corrective action program (Condition Report (CR)# 00061124). This is being treated as a Non-Cited Violation.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. **Key Points of Contact**

Exelon Generation Company

J. T. Anthony, Maintenance Director
P. Davison, Site Engineering Director
M. Delowery, Senior Manager-Outages
J. Doering, Site Vice President
E. Eilola, Shift Operations Superintendent
C. Hardee, Supervisor Emergency Preparedness
G. Johnston, Plant Manger
H. Trimble, Radiation Protection Manager
D. Warfel, Senior Manager, Design Engineering
A. Winter, Manager, Regulatory Assurance

b. List of Items Opened, Closed, and Discussed

Opened

None

Closed

50-277;278/00-08-01	URI	Previous Operability Concerns with Emergency Diesel Generators due to Heat Exchanger Crossflow (10 CFR Part 21 Issue).
2-00-002	LER	Emergency Diesel Generators Being in a Degraded Condition Outside of the Design Basis for Operation, in a Condition Prohibited by Technical Specifications, and in a Condition Which Could Have Prevented the Fulfillment of the Safety Functions of the Diesel Generators.
2-01-001	LER	Loss of Offsite Power Source Results in Specified System Actuation and Safety System Functional Failure.

Opened /Closed

50-277;278/01-06-01	NCV	Units 2 and 3 High Pressure Coolant Injection Suction from the Torus Check Valves (CHK-2-23B-61 and CHK-3-23B-61) Not Tested per ASME

		Section XI Inservice Testing (IST) Requirements. (Section 1R13)
50-277;278/01-06-02	NCV	Emergency Diesel Generators Being in a Condition Outside the Design Basis for Operation. (Section 4OA7)
50-277;278/01-06-03	NCV	Procedural Inadequacies with SO 53.7.D, "Response to a Loss of #343 Off-Site Startup Source" Identified During the Loss of One Offsite Power Source. (Section 4OA7)

c. Documents Reviewed

Plant Modifications

ECR 98-00836	SRV Discharge 3/4" VRV, 10 CFR 50.59 and Documentation Changes, April 8, 1998.
ECR 98-02864	Unit 3 SRV Discharge 3/4" VRV Removal-3R12, November 9, 1998.
ECR 99-00095	Modification to HPSW/ESW Ventilation, January 11, 1999.
ECR 99-00215	Change Unit 3 HWC System Controls to Correspond to 0 to 20 SCFM, January 27, 1999.
ECR 00-00763	Unit 3 Outboard MSIV Steam Leak Based on Sound When Door's Open, May 9, 2000.
ECR 00-01874	Upgrade of Unit 3 Service Water Valves, December 26, 2000.
ECR 98-02758	P00507 Reactor Stability - Power Range Modification - Unit 3, October, 27, 1998.
ECR 99-02624	E13 4KV Bus Undervoltage Relay Replacement- 3R13, November 24, 1999.
ECR 98-01304	E23 4KV Bus Undervoltage Relay Replacement- 3R12, May 27, 1998.
ECR 99-02547	ECR Required to Put Diodes Across MSIV Solenoids U/3, November 12, 1999.
ECR 99-02131	RWCU Inboard Isolation Valve (MO-3-12-015) - Wiring Changes to Operate the Valve Under Limit Control, September 28, 1999.
ECR 99-01506	Replace MO 3-12-015 with Gate Valve, June 25, 1999.
ECR 00-01726	Add EDG Keep Warm System Isolation Valve, November 22, 2000.

10 CFR 50.59 Safety Evaluations

ECR 9900015	Power Range Monitor System Replacement/Reactor Stability Modification, January 31, 2001.
ECR 98-00836	SRV Discharge 3/4" VRV, 10CFR 50.59 and Documentation Changes, April 8, 1998.
ECR 98-02244 & ECR 98-02245	Engineering Analysis PEAM-0003, Modification to High Pressure Service Water/Emergency Service Water Ventilation.
ECR 01-00329	Electrical Back-seating of MO-3-12-015, March 30, 2001.

PB-2001-0021	Evaluate/Delete Open Safety Function of Vacuum Relief Valve for HPCI Exhaust Drain Line, March 29, 2001.
PB-2001-0067	RBCCW Radiation Monitor Acceptance Criteria Change, May 21, 2001.
PB-2001-0147	HPSW Piping Minimum Wall Problem, May 25, 2001.

10CFR50.59 Safety Screens

ECR 9902547	ECRs to Install Diodes in the MSIV DC Solenoid Circuit (99-02547 and 99-02562), November 17, 1999.
ER 0100310	Replacement of Magnetic Only with Thermal-magnetic Breaker for E324-O-A, May 14, 2001.
ECR 99-00215	Change Unit 3 HWC System Controls to Correspond to 0 to 20 SCFM, Revision 0.
ECR 00-01874	Upgrade of Unit 3 Service Water Valves, Revision 0.
ECR 00-00763	Unit 3 Outboard MSIV Steam Leak Based on Sound When Door's Open, Revision 3.
PB-2001-0003-S	Procedure SO 5.7.B.C-2 COL, C Feedwater Heater String For Long Path Recirc, March 20, 2001, Revision 9.
PB-2001-0149-S	Procedure AO 30B.1, Alternate Cooling Water Supply To The Service Water Pumps And Return To Normal, June 29, 2001, Revision 0.
PB-2001-0042-S	Evaluate P/N Change For 6 Crosby Relief Valves, March 28, 2001, Revision 0.
PB-2001-0020	Electrical Back-seating of MO-3-12-015, March 30, 2001.
PB-2001-0021	Evaluate/Delete Open Safety Function of Vacuum Relief Valve for HPCI Exhaust Drain Line, March 29, 2001.
PB-2001-0067	RBCCW Radiation Monitor Acceptance Criteria Change, May 21, 2001
PB-2001-0147	HPSW Piping Minimum Wall Problem, May 25, 2001.

Performance Enhancement Program (PEP) Reports

I0009546	Error in 50.59 for 2C RHR Heat Exchanger FME, March 9, 1999
I0009846	Failure of Outboard MSIV DC Solenoid, May 13, 1999
I0009980	Inadequacies with 50.59 Reviews Supporting UFSAR Changes, June 1, 1999.
I0011035	Human Performance/Organization Issues Impact Design Change Quality, April 6, 2000.
I0011529	EDG Cross Flow Issue, May 2, 2000.
I0012359	EDG Jacket Water Coolant Check Valve Spring Broken/Foreign Material, March 10, 2001.

Non-Conformance Reports

01-00593	Check Valve 52E-10083A Broken Spring, June 5, 2001.
00-01129	Nonconformance on EDG Cross Flow Issue, August 4, 2000.

Procedures

LR-C-10	Performance Enhancement Program (PEP), Revision 11, Dated April 3, 2000.

LR-CG-10	Performance Enhancement Program (used to implement the PEP), Revision 4, Dated April 3, 2000
MOD-C-9	Design Control and Processing of Engineering Change Requests (ECRs), Revision 12.
LR-C-13	10 CFR 50.59 Reviews, Revision 9, Dated March 13, 2001.
LR-CG-13	Performing 10 CFR 50.59 Reviews," Revision 4, Dated March 13, 2001.
LR-CG-13-2	10 CFR 50.59 Review Determination Checklist, Revision 4, Dated March 13, 2001.
LR-CG-13-3	10 CFR 50.59 Screening," Revision 6, Dated March 13, 2001.

Drawings

6280-M-353	Reactor Recirculation Pump System P&ID, Revision 44, Sheets 1 and 2.

Calculations

PM-677	Emergency Diesel Generator Operability Curves for Various ESW Flows and Temperatures, Revision B.
PM-678	Performance Curves for Emergency Diesel Generator Heat Exchangers to Support Generic Letter 89-13 Monitoring Program, Revision B.
PM-1042	Determination of Diesel Operability with Cross-flow, Revision 1.