U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket/Report No.	50-277/90-25 50-278/90-25	License Nos. DPR-44 DPR-56
Licensee:	Philadelphia Electric Company Peach Bottom Atomic Power Station P. O. Box 195 Wayne, PA 19087-0195	
Facility Name:	Peach Bottom Atomic Power Station Units 2 and 3	
Dates:	December 4 - January 7, 1991	
Inspectors:	J. J. Lyash, Senior Resident Inspector R. J. Urban, Resident Inspector L. E. Myers, Resident Inspector	

Approved By: L. T. Doerflein — 1/15/91

L. T. Doerflein, Chief — Date
Reactor Projects Section 2B
Division of Reactor Projects

Areas Inspected:

The inspection included routine, on-site regular, backshift and deep backshift review of accessible portions of Units 2 and 3. The inspectors reviewed operational safety, radiation protection, physical security, control room activities, licensee events, surveillance testing, engineering and technical support activities, and maintenance.

TABLE OF CONTENTS

EXECUTIVE SUMMARY
Peach Bottom Atomic Power Station
Inspection Report 90-25

Plant Operations

Observation of a non-licensed operator implementing the daily plant tour and surveillance indicated that he was knowledgeable of his duties, procedures, and equipment; well trained in assessing plant and equipment conditions; and demonstrated good initiative in evaluating plant equipment status beyond the minimum required by the operator round sheets (Section 1.3).

During the report period both units remained at or near full power and only minor events or problems occurred which were well handled by the staff.

Maintenance and Surveillance

Licensee Administrative Procedures are not consistent with respect to the definition and implementation of independent verification (I.V.). This lack of clarity was evident in discussion with technicians in the field. The inspector observed that technicians did not perform adequate I.V. during an ST. The licensee was aware of the weakness, and was implementing corrective action. Because the incident was of minor safety significance, and the licensee is implementing corrective action, a Notice of Violation will not be issued (Section 3.0, NON 90-25-02).

Radiation Controls

Investigation and technical evaluation of an anomalous pure beta dose to an individual's personal dosimeter was thorough and adequate (Section 5.2).

Assurance of Quality

An Event Investigation Report reviewed by the inspector addressed an incident in which a water tight door had been improperly blocked open, but did not identify that the door was also a fire door. As a result, the specified corrective actions were not adequate. In response to the inspector's concern, the licensee is investigating the occurrence (Section 1.4, UNR 90-25-01).

The inspector identified a case in which the revision of a surveillance procedure was not effectively controlled, resulting in its issuance with an incorrect acceptance criterion (Section 3.0, UNR 90-25-03).

DETAILS

1.0 PLANT OPERATIONS REVIEW (71707, 93702)

The inspector completed NRC Inspection Procedure 71707, "Operational Safety Verification," by directly observing activities and equipment, touring the facility, interviewing and discussing items with licensee personnel, independently verifying safety system status and limiting conditions for operation, reviewing corrective actions, and examining facility records and logs. The inspectors performed 65 total hours of on-site backshift inspection, including 7 hours of deep backshift and weekend tours of the facility.

1.1 Philadelphia Electric Company Management Changes

On December 21, 1990, Philadelphia Electric Company announced that Mr. John Madara, General Manager of Nuclear Quality Assurance, had been elected an Officer of the Company. Mr. Madara will assume the position of Vice President, Production, replacing Mr. John Kemper who will retire at the end of December. Mr. John Cotton, currently Peach Bottom Atomic Power Station (PBAPS) Operations Superintendent, will replace Mr. Madara as General Manager of Nuclear Quality Assurance. Mr. Thomas Niessen, currently a PBAPS Shift Manager, will replace Mr. Cotton as PBAPS Operations Superintendent. Also it was announced that Mr. Kenneth Powers will be transferred to the PBAPS staff in February to begin a familiarization program leading to his becoming the Plant Manager. Mr. Powers will replace the current Plant Manager, Mr. John Franz, who will retire in the fourth quarter of 1991.

1.2 Operational Overview

Unit 2 began the inspection period at full power. Reactor power was reduced briefly to about 75% on December 10, 15, and 29 for control rod pattern adjustment, then returned to full power. On December 24, power was reduced briefly due to reduced grid requirements.

Unit 3 began the inspection period at 100% power. Power was reduced to 50% on December 21 for a control rod pattern exchange and to perform corrective maintenance on the #1 main turbine control valve and a leaking main steam isolation valve. Power was returned to 100% on December 24 and remained there through the end of the inspection period.

A detailed chronology of plant events occurring during the inspection period is included in Attachment I.

1.3 Non-Licensed Operator Rounds

The objective of this inspection was to evaluate the performance of a non-licensed operator. Areas evaluated were responsiveness to various plant parameters and conditions, adherence to procedures, documentation and communication of equipment status changes, and effective monitoring of plant equipment.

On January 4, 1991, the inspector accompanied an auxiliary plant operator (APO, non-licensed) on his daily rounds. Items checked by the APO included general equipment conditions, lubrication levels, gas bottle pressures, instrument temperatures, reactor water levels, and housekeeping. The individual acceptably answered various questions asked by the inspector. The APO described what he looked and listened for as he passed various operating equipment in the plant. Prior to performing certain equipment operations, the APO contacted the control room and notified operators clearly and exactly as to what he was about to do. The APO was very knowledgeable of his duties and procedures, and displayed a commendable attitude towards his work.

1.4 Event Investigation/Reportability Evaluation Form Review

The licensee established the Event Investigation/Reportability Evaluation Form (EI/REF) system about one year ago to provide a vehicle for follow-up investigation, root cause determination, and corrective action development for plant events or deficiencies. The inspectors routinely review a sample of completed EI/REFs. During the period the inspector reviewed EI Report 2-90-161. In November 1990, craft personnel blocked open a water-tight door between the Unit 2 'B' residual heat removal (RHR) pump room and the torus room. The door was blocked open to route welding machine power cables through the opening. The cables were not removed and the door was not closed when the workers left for the night. An operator identified the open door, the cables were removed and the door was closed. Procedure A-134, "Flood Protection Program for ECCS Systems," describes the controls required for blocking open a water tight door. It requires that the worker notify security (due to the resultant alarm), and that the open door be attended. Licensee corrective action was to provide training to craft personnel concerning the content of procedure A-134.

The inspector noted that this door penetrates a 3-hour rated fire barrier and is also considered a fire door. Procedure A-12.1, "Administrative Controls and Compensatory Actions Required for Fire Protection Impairments," requires that personnel complete a Fire Protection System/Feature Impairment Evaluation Form prior to impacting the integrity of fire protection equipment. This ensures adequate tracking, review, approval and compensatory actions in response to planned impairments. None was initiated in this case. In response to the inspector's concern, the licensee reviewed the fire suppression and detection equipment operability history and fire watch assignment records for the area. A separate impairment had prompted implementation of an adequate fire watch in the area during this time. Licensee review of other areas affected by the same work activity raised questions regarding the possible blocking open of the 'A' RHR pump room door. The licensee was continuing the investigation at the close of the inspection period. The EI initially performed in response to the incident was inadequate in that it had not identified this issue. The inspector also questioned the level of control applied to blocking water-tight doors that serve as flood barriers. The program as described in A-134 does not require pre-approval, informing operations, or provide for any tracking mechanism. The licensee indicated that the following actions would be taken:

- o investigate if other doors used during the activity were blocked open, and if adequate compensatory measures were in place;

- o investigate the reasons for the craft failure to utilize the approval process defined in Procesure A-12.1 prior to disabling the fire door;

- o evaluate the adequacy of water tight door controls as described in Procedure A-134, and

- o implement corrective actions to address the results of the reviews discussed above.

This item will remain unresolved pending review of the licensee's investigation results and corrective actions (UNR 90-25-001).

2.0 FOLLOW-UP OF PLANT EVENTS (93702, 90712, 92700)

During the report period the inspectors evaluated licensee staff and management response to plant events to verify that root causes were identified and appropriate corrective actions implemented. During the inspection period both units continued to operate at or near full power with only minor plant problems and events. These events were handled well by the licensee staff. The inspectors also observed licensee preparation for and conduct of significant operational evolutions such as removal from and return of systems to service, and reactor power changes. These evolutions were well planned and controlled.

3.0 SURVEILLANCE TESTING OBSERVATIONS (61726, 71707)

The inspectors observed surveillance tests to verify that testing had been properly scheduled, approved by shift supervision, control room operators were knowledgeable regarding testing in progress, approved procedures were being used, redundant systems or components were available for service as required, test instrumentation was calibrated, work was performed by qualified personnel, and test acceptance criteria were met. Daily surveillances including instrument channel checks, jet pump operability, and control rod operability were verified to be adequately performed. The following tests were also observed during the inspection period, with no significant questions or concerns identified by the inspector:

- o SI3A-02-ECCS-B1FM, "Functional Test of ECCS B/D-1 Card File";

- o SI3P-13-87-A1FM, "Functional Test of RCIC Low Steam Pressure Instrument PS 2-13-87B";

- o SP 1368, "Core Spray Motor Oil Cooler Test";

o ST 6.10.1-2, "Containment Cooling Systems Operability", and

o ST 6.11-2, "RCIC Pump, Valve, Flow & Cooler."

Review of two additional surveillance activities by the inspector identified issues requiring licensee attention. These observations and their resolution are discussed in detail below:

o ST 1.1, "HPCI Logic System Functional Test."

On December 18, 1990, the inspector observed two I&C technicians performing ST 1.1, "HPCI Logic System Functional Test." The inspector noted that one of the I&C technicians directly involved in performing the ST signed an independent verification (I.V.) step in the same manner as a double verification (D.V.) step. The inspector questioned the I&C technicians concerning the difference between D.V. and I.V. Both stated that they were the same. The I&C foreman also gave the same answer and added that informal training on this subject had just been given at all hands meetings the week of December 3, 1990.

The inspector obtained a memo describing the informal training referenced by the technicians. The intent of the training was to elevate the definition and conduct of D.V. to be equivalent to I.V. Apparently, I&C technicians did not interpret the training in this manner.

The application and definition of D.V. and I.V. are addressed in several licensee procedures including: Nuclear Group Administrative Procedure (NGAP) NAO3V001, "Verification"; Administrative Procedure A-47, "Surveillance Test Procedure"; and the Operations Management Manual. These documents are not consistent in their description. However, A-47 does state that the person performing the I.V. can not be directly involved with the initial performance of the task. In the example witnessed by the inspector, the I&C technician who performed the I.V. was directly involved in the initial task performance.

Corrective action taken by the licensee during the inspection period consisted of redefining D.V. and I.V. and re-training I&C technicians during all hands meetings. The licensee committed to develop and conduct formal training on D.V. and I.V., and to revise applicable Administrative Procedures for to be consistent. During the remaining part of the inspection period, the inspector noted proper performance of D.V. and I.V. by I&C technicians in all cases.

In the specific example identified by the inspector the procedure step was appropriately completed and a second verification performed, although not in accordance with the definition of I.V. The safety significance of this deficiency was minor. The licensee was aware of this general weakness prior to the inspector's observation, and was working to resolve it. Proposed licensee corrective actions appear adequate. The violation will not be cited because the criteria specified in Section V.A. of the Enforcement Policy were satisfied. This item will remain open pending review of the effectiveness of the planned licensee corrective actions (NON 90-25-02).

o ST 8.7 "Emergency Transformer Daily Surveillance."

The 4160/480 VAC load center transformers at Peach Bottom are gas cooled. In the past the licensee experienced problems with gas leakage, resulting in elevated hot spot temperatures and reduced transformer performance. The identified leaks were repaired and ST 8.7 was implemented to monitor gas pressures and transformer temperatures. The inspector reviewed the engineering evaluations and vendor information that establish the minimum acceptable gas pressures, leakage rates, and temperatures. ST 8.7 includes daily verification that gas pressures and temperatures are acceptable. Because gas pressures vary with transformer load, no leak rate acceptance criterion is included in the procedure. The system engineer performs an ongoing assessment of leakage rate as part of the Plant Performance Monitoring Program, through review of ST results. However, because system engineer review of the ST results may not be frequent enough to promptly identify a leak, the licensee indicated that the maintenance procedure used for recharging the transformers would be revised to require system engineer notification if the recharge rate exceeds once per 60 days.

The inspector noted that ST 8.7, Revision 7, also requires verification that transformer temperature indication does not exceed 200 degrees Celsius (C). This is an appropriate limit for hot spot temperature. Field walkdowns by the inspector indicated that two of the instruments were replaced with skin temperature indicators. The maximum value for skin temperature should be about 100 degrees C. The licensee agreed and revised the procedure. During further review the inspector noted that this same change had previously been made by the licensee, and was reflected in Revision 6. In making unrelated changes during the processing of Revision 7, the previous correction of the temperature acceptance criterion was somehow undone. The procedure mark-up approved by the Plant Operations Review Committee and forwarded to Nuclear Records was correct. Post-revision technical review by the sponsor of the revision focused only on those areas changed. The cause of the error had not been determined before the close of the inspection period. In this case the resulting error was minor. However, if a general underlying weakness exists it could cause more serious problems if left uncorrected. The cause for this error is still under investigation by the licensee. This item will remain unresolved pending review of the licensee's investigation results (UNR 90-25-003).

4.0 MAINTENANCE ACTIVITY OBSERVATIONS (62703, 71707)

The inspectors reviewed administrative controls and associated documentation, and observed portions of ongoing work. Administrative controls checked included blocking permits, fire watches and ignition source controls, QA/QC involvement, radiological controls, plant conditions, TS LCOs, equipment alignment and turnover information, post-maintenance testing and reportability. Documents reviewed included maintenance procedures, item handling reports, radiation work permits, material certifications, and receipt inspections. No concerns were identified during these reviews.

5.0 RADIOLOGICAL CONTROLS (71707)

5.1 Routine Observations

During the report period, the inspector examined work in progress in both units and included health physics procedures and controls, ALARA implementation, dosimetry and badging, protective clothing use, adherence to RWP requirements, radiation surveys, radiation protection instrument use, and handling of potentially contaminated equipment and materials.

The inspector observed individuals frisking in accordance with HP procedures. A sampling of high radiation area doors were verified to be locked as required. Compliance with RWP requirements was verified during each tour. RWP line entries were reviewed to verify that personnel had provided the required information and people working in RWP areas were observed to be meeting the applicable requirements. No unacceptable conditions were identified.

5.2 Anomalous Personal Dosimeter Reading

On August 22, 1990, the licensee informed the Resident Inspector that a personal thermoluminescent dosimeter (TLD) badge issued June 1, 1990, and measured on July 15, 1990, and found to have a beta skin dose of 2,784 millirem (mrem) and a whole body dose of 38 mrem. The exposure of the TLD indicated virtually a pure beta source. The individual's self reading dosimeter record indicated a dose of 20 milliroentgen for the same time period.

Since a pure beta source is not characteristic of the isotopic spectrum in any area of the plant, the licensee examined instrument calibration sources that could yield similar results. A 2 millicurie strontium/yttrium (SR/Y-90) check source, a pure beta emitter, is used in the health physics instrument cage on the 116 foot elevation of the turbine building to source check portable survey instruments. This source is accessible to plant personnel. The SR/Y-90 source is housed in an enclosure with appropriate double interlocks so that inadvertent exposure is not a concern.

To determine if the check source was the source of the TLD exposure, the licensee placed several TLDs in different locations within the source enclosure and exposed the TLDs for varying times. The doses measured on the TLDs were plotted on curves of dose versus time for beta and gamma exposures. The beta:gamma ratios were identical to those exhibited by the personal TLD and indicated that the TLD was probably exposed to the check source for about 140 seconds. The licensee then initiated an investigation to determine the circumstances leading to the exposure.

During September 1990 the inspector reviewed the individual's TLD dose values, the beta-gamma dose curves and other supporting documents, and discussed the results with plant personnel. The inspector concurred with the licensee's conclusion that the individual's TLD was exposed to the SR/Y-90 check source, and that it was not the result of an exposure of the individual to a beta source. The inspector had no further questions.

6.0 PHYSICAL SECURITY (71707)

The inspector monitored security activities for compliance with the accepted Security Plan and associated implementing procedures, including: security staffing, operations of the CAS and SAS, checks of vehicles to verify proper control, observation of protected area access control and badging procedures on each shift, inspection of protected and vital area barriers, checks on control of vital area access, escort procedures, checks of detection and assessment aids, and compensatory measures. No inadequacies were identified.

7.0 INSPECTION OF PEACH BOTTOM UNIT 1 (83726, 81070)

During the period the inspector reviewed the surveillance program applied to Peach Bottom Unit 1. On April 25, 1990, NRC issued to Philadelphia Electric Company Amendment No. 7 to this Possession-Only License No. DPR-12 for Peach Bottom Unit 1. This Amendment renewed the license until December 24, 2015, and revised the Technical Specifications (TS).

Peach Bottom Unit 1 was a high temperature gas cooled reactor that was operated from June 1967 to final shutdown on October 31, 1974. The plant was retired and placed in Safe Storage (SAFSTOR). SAFSTOR is the status of a facility that is placed and maintained in a condition that allows it to be safely stored, and subsequently decontaminated to levels that permit release of the facility for unrestricted use after decay of the activation and fission products.

The reactor vessel, primary system piping and steam generators remained in place. Except for electrical insulation and graphite components within the reactor vessel, all flammable materials have been removed. These included all charcoal traps from the helium purification system, all oils and other flammable liquids and solids. All radioactive liquids and the liquid waste system have been removed, and refrigerants and cooling water have been drained.

Access to high radiation areas is prevented with multiple bolted or welded barricades. Access to clean inspection areas is provided through locked gates or doors. Provision has been made for visual inspection of the accessible areas, including the subpile room and the containment sump. All penetrations to the containment are capped. A ventilation filter is installed in the equipment hatch for atmospheric pressure equalization of the containment vessel and as a check for the presence of airborne radioactive materials.

Residual radionuclides at Unit 1 consist of activation products in the reactor vessel and its internals (99% of the total activity), fission product contamination in the cooling system and contamination of the reactor building with activation and fission products. The TS require control of the residual radionuclides by access control, surveillance of ground water intrusion into the buildings and sumps, and periodic monitoring of the buildings for radiation levels, contamination, and airborne activity. Monthly surveillances were established to monitor accessible areas below ground level for water accumulation. A semi-annual surveillance was established to insure integrity of barriers and locks, and to expand radiological surveys. The licensee implements the monthly surveillance by procedure ST 12.12, "Peach Bottom Unit 1 Inspection for Water Intrusion," Revision 0, and the Semi-annual surveillance by procedure ST 12.12.1, "Peach Bottom Unit 1 Exclusion Inspection."

The inspector reviewed the amended license, the revised TS, and surveillance procedures to verify that all requirements were being adequately implemented. In addition, the inspector accompanied licensee personnel while performing ST 12.12.1. No evidence of ground water intrusion into the building was observed. All sumps were dry except one, which was properly sampled. The inspector also reviewed the results of ST 12.12 performed on December 18, 1990. The inspector found no discrepancies and had no further questions.

8.0 PREVIOUS INSPECTION ITEM UPDATE (92702, 92700, TI 2515/65)

(Closed) 50-277/87-12-001, Acceptability of Reverse Direction Local Leak Rate Testing of Certain Containment Isolation Valves (Common).

The licensee performs leak rate tests on numerous gate, globe, and butterfly valves in the reverse direction; that is, in the opposite direction to which the valve would be required to perform its safety function. 10 CFR 50, Appendix J, paragraph III.C.1, requires that containment isolation valves be leak tested in the positive direction, unless it can be demonstrated that reverse direction leak rate testing yields equivalent or more conservative results.

The licensee submitted a request to NRR for exemption from the requirements of 10 CFR 50, Appendix J, III.C.1, for identified inboard containment isolation valves to allow for reverse direction testing. NRR found the exemption unnecessary since the licensee justified that equivalent leakage measurements will result from applying the test pressure in either direction. This justification was found to be adequate for all valves discussed. The inspector noted that MO-10-31A/B, HPCI test line, Units 2 and 3, where not listed in the submittal, although valves of similar design discussed. In response to the inspector's question the licensee provided the justification for reverse direction testing on MO-10-31A/B, a solid wedge gate valve, which is symmetrical about the disc and seat. The stem force holding the disc to the seating surface is more than seven times the force exerted to open the valve by peak post-accident containment pressure of 49.1 psig. The sealing capabilities of this valve are the same regardless of the direction in which the pressure is applied. The reverse direction testing is acceptable based upon the information reviewed. The inspector had no further questions.

(Closed) UNR 87-25-002, Scram Discharge Volume Integrated Testing Requirement (Common).

During NRC Inspection 87-25 the inspector questioned the need to perform a periodic integrated scram discharge volume (SDV) test. On July 7, 1980, the NRC sent a letter to all operating BWRs. The letter requested Technical Specification (TS) changes within 90 days to provide limiting conditions for operation and surveillance requirements for SDV vent and drain valves, and reactor protection system and control rod block SDV level switches. Enclosed with the letter were model TS that would assist the utility in preparing their submittal.

In response to the above letter, the licensee submitted proposed TS changes dated October 14, 1980. The NRC approved the TS Amendments for Units 2 and 3 on March 1, 1983. On June 24, 1983, the NRC issued a Confirmatory Order for Unit 3 requiring permanent SDV modifications and applicable TS covering the modified system. The Unit 2 modification was complete in July 1982. However, the Confirmatory Order enclosed a newer version of model TS for guidance. This version of the model TS had the provision to test the operability of the entire scram system during each operating cycle. Scram instrument volume response and valve function at rated pressure and temperature were to be demonstrated by scramming the plant from approximately 50% control rod density. In an August 22, 1983, letter to the NRC, the licensee stated that there were differences between the model TS received in 1980 and 1983. The licensee stated that the dissimilarities were minor, and that they would not be submitting additional TS changes. No correspondence was issued by the NRC in response to their letter.

Long-term corrective action to resolve generic concerns with the adequacy of SDV system designs was being tracked by NRC Multi-Plant Action (MPA) Item B-58. On April 25, 1990, the NRC issued Generic Letter 90-04, "Request for Information on the Status of Licensee Implementation of Generic Safety Issues Resolved with Imposition of Requirements or Corrective Actions." Item MPA B-58 was addressed in GL 90-04. A June 29, 1990, response from the licensee stated that actions were complete for MPA B-58. A November 6, 1990, response to the licensee acknowledged completion of MPA B-58. Based on the above correspondence, this open item is closed.

(Closed) Notice of Violation 89-16-003, HPCI Wiring Error Due to Successive Failures of Modification Installation, Inspection and Testing.

Two electric leads were reversed following implementation of a modification which installed an analog isolator in the high pressure coolant injection (HPCI) system. Several of the process barriers which should have identified and corrected the discrepancy were not effective. Weaknesses identified included: 1) the craft worker did not follow the installation instructions; 2) the Quality Control Inspector did not adequately verify compliance with the design drawings; 3) the Field Engineer did not adequately inspect the completed work, and 4) the modification acceptance test (MAT) used was the routine system surveillance test, which did not cover the scope of the modification. The licensee identified the problem when HPCI failed the 150 psig surveillance test run during the subsequent plant startup. Although this violation was identified by the licensee, the NRC issued a Notice of Violation (NOV) because of the multiple failures of the process to identify the problem prior to plant startup.

The licensee performed an investigation to determine the root causes, and documented the results in Licensee Event Report 50-277/89-009, and in their response to the NOV dated July 19, 1989. The details of the incident and the proper implementation of the installation and inspection instructions were discussed with the craft, Field Engineers and Quality Control Inspectors during routine training sessions. Applicable modification installation administrative procedures were revised to clarify the contents of Installation Checklists and inspection requirements. The pre-implementation review requirements for MATs have also been changed to ensure that they are appropriately reviewed for technical scope. The inspector reviewed these procedure revisions and training records. The inspector also noted that during the past two years the licensee has dedicated considerable effort to strengthening the modification process. These efforts will reduce the likelihood of problem recurrence. The inspector had no further questions.

(Closed) TMI Action Plan Item II.E.4.2(7), Installation of a High Radiation Isolation Signal for the Primary Containment Vent and Purge Line Isolation Valves.

Item II.E.4.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements," provided the results of the NRC staff's evaluation of features needed to improve containment isolation dependability. Sub-item number (7) required that containment purge and vent isolation valves close on a high radiation signal. Following negotiation with the NRC, the licensee installed a high radiation isolation signal on all containment vent and purge valves greater than two inches in diameter on both units. The valves automatically close if a main stack high-high radiation signal is received, a containment vent path exists (two in-series valves open) and there is flow through the standby gas treatment system (SGTS). The two main stack radiation sensors, the single SGTS flow sensor and the associated circuitry are not safety-related. The details of the design were submitted to the NRC for review and subsequently approved. The licensee also submitted a Technical Specification (TS) Change Request to incorporate operability and surveillance requirements for the main stack radiation monitor high-high radiation trip signal and logic. This submittal was reviewed and approved by the NRC, and issued as Amendment Numbers 156 and 158 to the TS for Unit 2 and Unit 3 respectively.

The inspector reviewed licensee and NRC correspondence related to this item, the completed modification package, electrical schematic and logic drawings, operating procedures and surveillance procedures. The modification was implemented as discussed in the licensee's submittals. Operation of the system is described appropriately in operations training materials and procedures. Surveillance test procedures have been established, scheduled and implemented as required by the revised TS. During the review the inspector noted that while operability and surveillance requirements for the main stack radiation monitors had been incorporated into the TS, no similar requirements addressing the SGTS flow sensor were established. Only a single sensor is present, and its failure would prevent operation of the isolation logic. The licensee provided procedure SI2F-9A-SBGT-XXOO, "Calibration Check of Standby Gas Treatment Exhaust Flow Instruments FT 20008, FS 20008 and FR 20008," which is performed once per operating cycle. This procedure performs the needed calibration and functional test of the flow switch. The inspector discussed the absence of TS requirements for this component with cognizant NRR personnel who indicated that the system design and TS were considered to be adequate. The inspector had no further questions.

9.0 MANAGEMENT MEETINGS (40500)

9.1 Routine Meetings

The Resident Inspectors provided a verbal summary of preliminary findings to the Peach Bottom Station Plant Manager at the conclusion of the inspection. During the inspection, the Resident Inspectors verbally notified licensee management concerning preliminary findings. No written inspection material was provided to the licensee during the inspection. This report does not contain proprietary information. The inspectors also attended the exit interview for the following inspection during the report period:

Dates	Subject	Report No.	Inspector
12/18-22	Water Chemistry/Inservice Inspection	90-23/23	Kaplan
12/18-21	Limited SRO - Fuel Handling	90-24/24	Pullani

9.2 Nuclear Review Board Meeting Attendance

On January 3, 1991, the inspector attended portions of a meeting of the licensee's Nuclear Review Board (NRB). The agenda prepared for the meeting was comprehensive. It included review of recent plant events, update of licensee staff actions in response to previously established NRB open items, discussion of NRC inspection report results and Notices of Violation, NRB Member plant tour observations and the results of other evaluations conducted by individual NRB Members. The agenda focused on safety significant issues. Discussion and questioning was detailed and frank. Of particular value was the overview of plant operations, and presentations describing staff follow-up to significant issues, provided by the Plant Manager at the opening of the meeting. These presentations provide the NRB with the opportunity to directly question the Plant Manager and the responsible staff regarding technical issues and the evaluation process applied to their resolution. The licensee's practice of using first line supervision and workers to present the issues provides direct feedback to NRB, and exposes the licensee staff to the broader assessment function provided by NRB.

ATTACHMENT I

Facility and Unit Status

Unit 2

12/4	Reactor at 100% power.
12/10	Reactor power reduced to 75% for control rod pattern change and returned to full power.
12/15	Reactor power reduced to 75% for control rod pattern change and returned to full power.
12/24	Reactor power reduced to 75% in response to reduced grid requirements. Returned to full power in 12 hours.
12/29	Reactor power reduced to 75% power for control rod pattern change and returned to full power.
12/30-1/7	Reactor at 100% power through the end of the period.

Unit 3

12/4-12/20	Reactor at 100% power.
12/21	Reactor power reduced to 50% for control rod exchange and corrective maintenance on the #1 main turbine control valve and a leaking main steam isolation valve.
12/24-1/7	Reactor power remained at 100% through the end of the period.