U. S. NUCLEAR REGULATORY COMMISSION
REGION 1

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License No.	DPR-35	Category C	50293-092384 50293-092884
Licensee:	Boston Edison Company 800 Boylston Street Boston, Massachusetts 02199		50293-092984

Facility Name: Pilgrim Nuclear Power Station

Dates: August 28, 1984 - October 8, 1984

Inspectors: Jon R Johnson 11/15/84
J. Johnson, Sr. Resident Inspector date

M McBride 11/15/84
M. McBride, Resident Inspector date

Approved By: L. Tripp 11/20/84
L. Tripp, Chief, Reactor Projects date
Section No. 3A, Projects Branch No. 3

Inspection Summary:
Inspection on August 28, 1984 - October 8, 1984 (Report No. 50-293/84-26)

Areas Inspected: Routine unannounced safety inspection of plant operations including followup of previous findings, operational safety verifications, followup on plant events and LERs, a review of surveillance and maintenance activities, IEB 84-03 followup and a review of quality assurance program procedural controls. The inspection involved 261 hours by two resident inspectors.

Results: One Violation was identified (Failure to properly review and approve quality assurance program-related procedures, Paragraph 8).

DETAILS

1. Persons Contacted

   Within this report period, interviews and discussions were conducted with members of the licensee (and contractor) staff and management to obtain the necessary information pertinent to the subjects being inspected.

2. Followup on Previous Inspection Findings

   a. (Open) Unresolved Item (82-24-02). Manual containment isolation valve criteria. Previous review of this item is discussed in NRC Report Nos. 82-30 and 83-14. During this period, the inspector reviewed memo NED 83-435, dated July 29, 1983, which summarizes the nuclear engineering department manager's evaluation of certain manual isolation valve criteria (regarding IEB 79-08 and TMI TAP item II.E.4.2). This evaluation concludes that isolation capabilities of the present design are adequate but that clarification and/or revision of the licensee's commitment should be provided to the NRC:NRR. The inspector met with a member of the licensing staff who has initiated this revision. This item remains open pending a review of the licensee's revised response.

   b. (Closed) Follow Items (82-27-03 and 82-27-04). Review implementation of internal exposure monitoring program commitments. Procedure 6.2-161, "Administration of the Internal Exposure Monitoring Program", was revised to incorporate the commitments made during Inspection 82-27. The internal exposure monitoring program was also reviewed during NRC Inspections 84-14 and 84-25. These items are closed.

   c. (Closed) Follow Item (83-08-02). Review revisions to four radiological environmental monitoring program (REMP) procedures. The four REMP procedures were suitably revised by July 1, 1983. This item is closed.

3. Operational Safety Verification

   a. Scope and Acceptance Criteria

      The inspector observed control room operations, reviewed selected logs and records, and held discussions with control room operators. The inspector reviewed the operability of safety related and radiation monitoring systems. Tours of the reactor building, turbine building, station yard, switchgear rooms, SAS, cable spreading room, battery rooms, HPCI room, diesel generator rooms, and control room were conducted. Observations included a review of equipment condition, security, housekeeping, radiological controls, and equipment control (tagging).

      These reviews were performed in order to verify conformance with the facility technical specifications and the licensee's procedures.

b. Findings

(1) On August 23, 1984, an NRC regional manager identified a space between the first cell of the "A" 125 volt batteries and the end of the battery rack. The licensee evaluated the effect of the space on battery availability and determined that the integrity of the batteries would not be jeopardized during a seismic event. However, the licensee is modifying the rack to eliminate the space. The evaluation is documented in licensee memo DM 84-206, dated September 14, 1984. The inspector had no further questions.

(2) On August 31, 1984, a licensee health physics technician stated that two small radioactive chips had been identified in the control rod drive repair room. The chips were similar to the chip that caused an unanticipated extremity radiation exposure on August 18, 1984. The chips were found during decontamination activities. On September 4, 1984, the licensee stated that a total of twelve chips had been identified in the room, including seven located in tool trays on a flush tank. A Region I specialist inspector followed up on the additional chips during NRC Inspection 50-293/84-25.

On September 5, 1984, the inspector attended a management meeting in Region I to discuss the unanticipated extremity exposure on August 18, 1984. The discussions at the meeting and licensee corrective actions are described in NRC Inspection Report 50-293/84-25.

(3) The inspector followed up on a reported "explosion" in the drywell at about 2:00 pm on August 31, 1984. A contractor working for the maintenance group on site was injured in the drywell when an inflatable plug slipped out of the feedwater piping and struck the man on the head and arm.

The inspector held discussions with personnel and reviewed records regarding this event. Personnel were in the process of performing a preliminary leak test on feedwater check valve No. 62A (outboard) following maintenance in accordance with procedure No. 3.M.4-51, Preliminary Leak Testing of Feedwater Check Valve with Test Cover, Rev. 1. The inflatable plug was installed in the feedwater piping inside the drywell in the upstream side of the disassembled 58A (inboard) check valve in order to provide a boundary to hold air pressure in the test volume for the 62A valve. During pressurization of the test volume the plug slipped out and struck the man who was watching it. The licensee performed first aid and decontamination of hair prior to transport to a local hospital for treatment. The licensee's investigation determined that, although the personnel were following approved station procedures, these procedures do not address any precautions regarding the use of an inflatable test plug. The licensee immediately suspended similar testing until adequate personnel safety precautions could be prescribed.

The inspector had no further questions regarding the licensee actions in response to this event.

(4) On September 12, 1984, the inspector attended Operations Review Committee (ORC) meeting no. 84-94. A quorum including the ORC chairman and three members (the Chief Maintenance Engineer, The Chief Radiological Engineer, and the Onsite Safety and Performance Group Leader) was present at the meeting. Plant design change packages, failure and malfunction reports, temporary modifications, proposed changes to the technical specifications, temporary procedures, and temporary procedure changes were reviewed during the meeting. An agenda was distributed to the ORC members prior to the meeting. The meeting lasted about three hours. The inspector had no further questions.

(5) The licensee's method of maintaining operability of the refueling floor ventilation system radiation monitors was reviewed. Four radiation monitors initiate secondary containment (one out of two twice logic) when radioactivity levels in the refuel floor exhaust duct are abnormally high.

Two of the four duct monitors are located on the 91 ft level of the reactor building, one floor below the refueling floor. The licensee stated that ventilation flow through the exhaust duct is required for the monitors to be operable. Licensee procedures do not, however, require that the exhaust flow be maintained. The refueling floor exhaust duct is normally used for ventilation but has been isolated during the outage for maintenance.

At the exit interview, the licensee stated that air flow must be maintained in the refuel floor ventilation ducts for the vent monitors to be operable. To ensure that this flow is maintained during fuel handling operations, the licensee stated that refueling checklists which require checks of the refuel floor vent monitors will be modified to incorporate flow rate checks. Also, procedure No. 2.2.54, "Refueling Floor Ventilation Exhaust Process Radiation Monitoring System", will be modified to incorporate technical specification action statements. The licensee stated that these changes would be made prior to the start of fuel handling activities.

(6) On September 27, 1984, the inspector reviewed the licensee actions regarding an alleged use of a controlled substance on site. This review included a tour and inspection of the turbine deck 51 foot elevation fan room, discussions with security and operations management personnel and a review of station records. On October 1, 1984, the licensee informed the inspector of the results of the investigation which included laboratory tests with negative findings.

The inspector had no further questions at this time. Observations of personnel conduct and the implementation of the licensee's policies and procedures will continue to be reviewed during routine inspections.

4. Followup on Events and Nonroutine Reports

a. Events

(1) On September 12, 1984, the licensee notified the resident inspector that the results of General Electric Co. etchings and x-ray fluorescence analysis confirmed the presence of about one inch of original (Combustion Engineering) 304 stainless steel furnace sensitized safe end material remaining on the A jet pump instrumentation reactor vessel nozzle.

Following a review of original construction records the licensee tentatively concluded that the unwanted section was inadvertently left attached because of a possible measurement error during shop fabrication prior to initial plant construction.

On September 23, 1984, intermittent circumferential ultrasonic indications were detected over a seven inch span on the safe end. No axial indications were detected. The indications did not go through wall. The licensee has installed a full-strength weld overlay on this nozzle, and at the conclusion of this inspection period was conducting baseline nondestructive examinations.

(2) On September 20, 1984 at 3:30 pm, the "B" diesel generator was inadvertently started during surveillance test 8.M.2-2.10.1-8, "Logic System Functional Test Core Spray System "A" Drywell High Pressure Auto Initiation Trip". The diesel started when an Instrument and Controls Technician accidentally closed relays 14A-K5B and 14A-K6B simultaneously. The relays should have been closed and opened in sequence during the surveillance.

The licensee suspended the surveillance and manually shut down the diesel generator. The technician was cautioned to more carefully follow the procedure before repeating the surveillance test. The inspector reviewed the surveillance records and noted that the procedural steps completed prior to the incident were repeated afterwards. The licensee notified the NRC via the ENS telephone line of the incident at 4:15 pm on September 20, 1984 and a Licensee Event Report was issued on October 10, 1984.

No violations were identified.

(3) On September 28, 1984 at 2:15 pm, a full scram signal was generated when the feed for a nonsafety 480 volt bus (B4) was shifted to a safety 480 volt bus (B2) from a nonsafety 4160 volt bus (A4). The

bus transfer was part of the isolation required for planned maintenance on the feeder breaker (B401) connecting B4 to A4. Bus B4 was transferred to B2 by closing a manual breaker (B410) between the two busses.

The licensee stated that closing breaker B410 between B2 and B4 generated a current spike that in turn tripped the feeder breakers to both busses (breakers B401, B201, and A608). A full scram signal was generated when the "B" RPS motor generator set lost power from bus B4 because the scram discharge volume high level trip was bypassed.

The licensee further stated that breaker B410 does not automatically open on a loss of power, shedding the nonsafety bus B4 from bus B2. This failure to open could prevent the safety bus from performing its safety functions. Safety bus B1 may also be connected to a nonsafety bus, B3, by manual breaker B310. The remaining 480 volt safety bus (B6) is powered by either B1 or B2 and can be indirectly connected to a nonsafety bus through B1 or B2.

The licensee currently has no procedure which controls the use of breakers B310 and B410. The licensee stated that the breakers were only used during outages for maintenance activities, when the safety busses were not required to be operable. At the exit interview, the licensee stated that instructions would be prepared containing guidance on when the breakers may be used and how the busses should be transferred (Open Item 84-26-01).

(4) At 3:51 am on September 29, 1984, the drywell was inadvertently sprayed with about 10,400 gallons of reactor cavity water from the A drywell spray header. As followup to this event, the inspector held discussions with licensee personnel, and reviewed operating logs and records of work in progress at the time.

The outboard drywell spray isolation valve (MOV 1001-23A) was being opened by electricians to check limitorque switch positions as a final part of approved preventive maintenance. Maintenance Requests (M.R.) 83-10-30 and 84-10-41 were approved and active for post work testing. Water was inadvertently sprayed because the other series valve (MOV 1001-26A) was already open. M.R. 83-10-30 was active for post work testing for this valve. The Residual Heat Removal (RHR) system was in the shutdown cooling and reactor cavity cleanup mode with D RHR pump discharging into the A recirculation loop and the fuel pool demineralizer system (and thus also pressurizing the drywell spray header).

Immediate corrective actions included stopping the operating RHR pump,manually closing MOV 1001-23A, and processing the water which collected in the drywell sump to radwaste tanks. The licensee also assembled teams to inspect for damage and held critiques to determine if further corrective actions were necessary. It was determined that, although no procedures had been violated, it was desirable to proceduralize a practice that had been used in the past but not in this instance, namely, for maintenance personnel to receive the control room operator's concurrence prior to cycling a valve for a post work test. Procedure 3.M.4-10 (Attachment C) was immediately changed to require the Watch Engineer to concur prior to conducting valve actuation tests.

No damage was observed but the licensee has completed a review and is tracking all safety related equipment that could have been damaged to ensure that adequate testing will be performed. Sections of wetted insulation are being replaced.

The inspector had no further questions at this time. No violations were identified.

b. Review of Licensee Event Reports (LERs)

LERs submitted to the NRC:Region I office were reviewed to verify that the details were clearly reported and that corrective actions were adequate. The inspector also determined whether generic implications were involved and if on site followup was warranted. The following reports were reviewed.

No.	Subject
84-10	CRD Collet Retainer Tube Defect
84-12	Unplanned start of the "B" Emergency Diesel Generator

The control rod drive retainer tube weld defects (LER 84-10) were reviewed during NRC inspections 50-293/84-17 and 84-23. At that time, the licensee stated that defects would be evaluated by General Electric Co. representatives. This evaluation had not been completed at the end of the current inspection and will be tracked under item 84-17-02.

The inspector reviewed the unplanned start of the "B" emergency diesel generator (LER 84-12) shortly after the incident. The inspector's review of this event is discussed in Section 4.a.(2).

The inspector noted that items tracked in the licensee's QA nonconformance system were not being reviewed for reportability to the NRC. The licensee stated that some nonconformances involved items, such as as-built defects in construction identified during the current outage, that may be reportable. The licensee has reviewed all open nonconformances which were initiated after January 1, 1984. No instances of improper reporting were identified. The licensee stated that future nonconformance reports would be routinely reviewed by the compliance group for reportability. The inspector had no further questions.

5. Surveillance Testing

a. The inspector reviewed the licensee's actions associated with surveillance testing in order to verify that the testing was performed in accordance with approved station procedures and the facility Technical Specifications.

Portions of the following tests were reviewed:

-- Low amplitude dynamic loading for seismic qualification studies of control room panels in accordance with procedure TP 84-166, Dynamic Response Testing of Control Room Panels, Revision 0, on August 30, 1984.

-- Routine calibration of the reactor water level indicator switches for the containment spray permissive logic (procedure No. 8.M.2-2.1.3) on September 13, 1984.

-- Routine logic system functional test of the core spray system "A" drywell high pressure auto initiation trip (procedure No. 8.M.2-2.10.1-8) on September 20, 1984.

b. Findings

(1) On September 13, 1984, the licensee notified the NRC via the ENS line that the as-found setting on the containment spray permissive water level switch (LITS 263-73 B) was out of tolerance in a nonconservative direction. The switch had been calibrated on October 9, 1983, to trip at a test input signal of 219 in. of water. The as-found trip setting on September 13, 1984 was 224 inches of water. Surveillance procedure No. 8.M.2-2.1.3. "Reactor Water Level Containment Spray Permissive", required that the Watch Engineer be notified if the as-found test input level is above 223 in. of water. The test input water level varies inversely with the water level in the core (i.e., increasing test input levels correspond to decreasing core water levels).

Table 3.2.B in the Technical Specifications requires that the containment spray permissive switches be set above two thirds core height (302 in. above vessel zero). Personnel in the site Operations, Technical and I&C Groups and in the corporate Engineering Group were unable to readily relate the test input levels to the technical specification limit.

The test input levels and core water levels are discussed in a corporate engineering review documented in internal memo I&Cs No. 82-41, "Pilgrim Station Reactor Water Level General Issues and Nuclear Engineering Recommendations", dated March 16, 1982. This document indicates that 224 in. of test input to the containment spray permissive level switches corresponds to a vessel water level of 312 in. above vessel zero (-40 in. indicated on the level switch output). Since the 224 in. test input level is ten inches above the technical specification limit (302 in. above vessel zero), the as-found setting on September 13, 1984, was in accordance with the Technical Specifications and did not have to be reported.

The following discrepancies were noted during the review of this incident:

-- Licensee documentation in the control room indicated that two thirds core coverage occurs at 307 in. above vessel zero, not at the 302 in. level stated in the technical specifications.

-- Plastic identification cards mounted next to the LI-106 A and B Yarway level instruments on panel 903 in the control room indicated that two thirds core coverage occurs at a value of -39 in. on the instruments. Other documentation in the control room indicated that two thirds core coverage (307 in. above vessel zero) occurs at -48 in. on the Yarway instruments.

-- Surveillance procedure 8.M.2-2.1.3 stated that two thirds core coverage occurs at 218 in. of test input. This input level corresponds to approximately 321 in. above vessel zero.

The licensee stated that an engineering review would be conducted to determine whether the two thirds core coverage limit in the technical specifications, 302 in. above vessel zero, needed to be revised. The licensee also stated that a maintenance request to change the two thirds core coverage level listed on the plastic cards on the 903 panel would be initiated. The inspector had no further questions. No violations were identified. This is being tracked as Open Item 84-26-02.

(2) Surveillance 8.M.2-2.10.1-8, "Logic System Functional Test Core Spray System "A" Drywell High Pressure Auto Initiation Trip", was reviewed in connection with the inadvertent start of the "B" emergency diesel generator on September 20, 1984. The licencee notified the NRC via the ENS line of the incident on September 20, 1984 at 1615 and subsequently reported the incident in LER 84-12. The inspector review of this incident is detailed in Section 4.a.(2) of this report.

6. Maintenance/Modification Activities

a. Scope

The inspector reviewed the licensee's actions associated with maintenance and modification activities to verify that they were conducted in accordance with station procedures and the facility Technical Specifications.

The inspector verified for selected items that the activity was properly authorized and that appropriate radiological controls, equipment control tagging, and fire protection were being implemented.

The items/documents reviewed included the following:

-- Inadvertent installation of 1 in. pipe which had been rejected by a testing laboratory in class I systems

-- Meeting to discuss the modifications of main steam line safety relief valves

-- Cutting of core spray piping outside containment

-- Plans to install a weld overlay on the "A" jet pump instrumentation nozzle (discussed in Section 4.a.(1))

-- Clearing of dye penetrant indications on control rod drive collet housings, and

-- Control of drilling in the reactor building floor during installation of enclosures around the motor control centers.

b. <u>Findings</u>

(1) On August 31, 1984, the licensee Q.A. Manager notified the inspector that their prime contractor (General Electric Company) was investigating the circumstances surrounding the receipt and installation of sections of 1 inch schedule 80 stainless steel piping which had been rejected by a subcontractor testing laboratory.

The inspector subsequently held discussions with licensee Q.A. department and G.E. representatives regarding the cause and corrective actions. The inspector reviewed the G.E. findings entitled, Report of Investigation - ASME Section III, Class 1, Small Bore Piping, Pilgrim Power Station, dated September 19, 1984. This report concludes that the cause of the problem was inadequate document review, receipt inspection and followup of reject remarks on subcontractor test reports and pipe markings (for which the subcontractors have initiated corrective actions).

The inspector verified that the G.E. purchase order (296N2272A01) required that quality records be supplied for the raw materials and stock along with fabricated piping sub-assemblies. These subcontractor quality records were received on site prior to installation and included ASME NPP-1 data forms and certificates of compliance.

The licensee stated that re-examination (ultrasonic testing) of the questionable sections was in progress but that a final decision on resolution to G.E. nonconformance report No. RS-001 had not been decided. Pending review of the licensee's resolution of acceptability of small bore piping installed for the recirculation piping replacement project, this item is unresolved (84-26-03).

(2) On September 20, 1984, the inspectors attended a meeting held onsite between the licensee maintenance and engineering representatives and their technical consultant from M.I.T. regarding changes made to the two stage Target Rock safety relief valves (SRVs). These changes were made as a result of a metallurgical analysis of the SRV setpoint drift identified during as found testing at an off site laboratory (see LER 84-05). The pilot valve disc material was changed from Stellite 6B to Stellite 21 to eliminate the presence of a large carbide phase and help prevent sticking of the pilot disc due to corrosion product buildup and mechanical interaction.

The licensee stated that the Target Rock Co., General Electric Co., and the BWROG have reviewed and accepted BECo's proposed resolution for this generic problem.

During the meeting, the licensee also stated that it was evaluating considerations that low pressure cyclying may have contributed to mechanical interaction and setpoint drift, and what effect this would have on T.S. surveillance testing requirements, if any.

The licensee also stated that an analysis of the effect of the as found condition of the SRVs was in progress and that the above corrective actions will be provided in an update report.

The inspector had no further questions. Additional findings regarding valve modifications will be documented in NRC Region I Inspection Report No. 50-293/84-31.

(3) On September 28, 1984, the inspector observed the cutting of a section of core spray piping adjacent to valve MOV 1400-25B. The piping was located outside of primary containment and was being cut to allow a sample of the weld (attaching valve MOV 1400-25B to the pipe) to be removed for metallographic analysis. The inspector reviewed General Electric Traveler PT 81-4, "Partial Pipe Replacement of Core Spray Outside Containment", Revision 1, dated September 26, 1984 which was used to control the work. No procedural discrepancies were identified. The workers were aware of procedural requirements and hold points. No violations were identified.

(4) The inspector reviewed a draft of a quality control group memo which described the clearing of penetrant indications on new style control rod drive collet housings. The memo indicated that a penetrant test of the entire surface of each collet was done to locate the indications. Following flapping and grinding, a second penetrant exam was conducted to verify that the indications were cleared. The

supervisor of the quality control group verified the accuracy of the memo. The inspector had no further questions. No violations were identified.

(5) On October 4, 1984, the inspector reviewed licensee controls over core boring into the reactor building floor. Holes for anchor bolts were bored into the floor during construction of enclosures around motor control centers. A potential violation for licensee control over the drilling activities was identified and documented in NRC Inspection Report 50-293/84-28.

7. Followup on NRC IE Bulletin 84-03

The inspector reviewed the licensee actions regarding NRC IEB No. 84-03, Refueling Cavity Water Seal, in order to determine whether the actions taken addressed the concerns identified.

This Bulletin described an event at another reactor plant involving gross failure of an inflatable pneumatic refueling cavity seal and requested that other licensees evaluate the potential for, and consequences of, a refueling cavity seal failure and provide a summary report to the NRC.

Following several discussions with licensee representatives who were preparing the licensee's response, the inspector questioned why the licensee's response had not been submitted yet since refueling was planned for the near future. Subsequently, the licensee submitted their report dated September 21, 1984.

The licensee's refueling seal is of a different design than that described in the Bulletin. The Pilgrim design consists of a permanent metallic bellows with a secondary spring seal to prevent gross leakage in the event of a failure of the primary seal.

The inspector also reviewed the Pilgrim Final Safety Analysis Report with regard to the refueling seal and reactor cavity/refueling pool design. FSAR Section 10.3 (Spent Fuel Storage) states that the spent fuel pool is a seam welded, reinforced concrete, class 1 structure. This section further states that, to avoid unintentional draining of the pool, there are no penetrations that would permit the pool to be drained below a safe storage level (about 10 ft above the fuel). FSAR figure 10.4-1, Fuel Pool Cooling and Demineralizer System also demonstrates seismic design boundaries and designs used to prevent siphoning of the pool.

The inspector had no further questions at this time. The licensee's response was determined to address the immediate concerns of the Bulletin.

8. Review of Quality Assurance Program - Related Procedures

Section 5 of the Boston Edison Quality Assurance Manual (BEQAM), "Instructions, Procedures, and Drawings", Revision 11, dated May 14, 1984 requires that selected quality assurance program-related procedures be reviewed and approved by the QA Manager. The criteria for determining which procedures are quality assurance program-related are not clearly outlined in the BEQAM. However, Section 5.2 in the BEQAM states that all Section 1 station procedures (administrative procedures) and certain Section 3 station procedures listed in BEQAM Exhibit II-5-1 (maintenance procedures) are QA program-related and shall be reviewed and aproved by the Quality Assurance Manager.

During a review of the station procedures, the inspector noted that approximately twenty-five procedures from Section 1 and four procedures from Section 3 were not reviewed and approved as required by the BEQAM. The unreviewed Section 3 procedures were:

-- Procedure 3.M.1-1, "Preventative Maintenance", dated May 13, 1983. (This procedure includes the instructions for calibration of major Instrument and Control components.)

-- Procedure 3.M.1-8, "Disposition of Nonconforming Materials", dated August 24, 1983.

-- Procedure 3.M.1-10.1, "Torque Wrench Calibration", dated June 16, 1980.

-- Procedure 3.M.1-10.3, "Calibration of Noncontrolled Lab Equipment", dated June 30, 1982.

The licensee stated that the Section 1 procedures which were not approved were not quality assurance program-related (e.g., procedures on key control and hard hat usage). However, other Section 1 procedures which were not quality assurance program-related (e.g., Oil Spill Prevention Control and Countermeasure Plan) were reviewed and approved by the QA Manager. The licensee had no clear explanation why some procedures were sent to the QA Manager for approval and others were not.

The licensee further stated that the Section 3 procedures were not sent to the QA Manager for review because they were to be revised in the near future. In response to the finding, the QA Manager issued a memo to the Plant Manager, requesting copies of three of the Section 3 procedures for review and approval. The fourth procedure, 3.M.1-1 was not requested for review because it was not considered quality assurance program-related.

The licensee stated that all new procedures are now sent to the QA department for review and that the procedures which should be approved by the QA Manager are listed in a document entitled "Index to QA Program Related Procedures to 10 CFR 50, Appendix B Criteria", Revision 0, dated April 13, 1984. However, at the close of the inspection, QA Manager had not reviewed and approved of all the procedures listed in that Index. The licensee subsequently modified Section 5 of the BEQAM to require that new procedures listed in the index (a controlled document) be approved by the QA Manager. This revision does not clearly require that current procedures listed in the Index to be approved by the QA Manager, only new procedures.

Failure to review and approve quality assurance program-related procedures as required by Section 5 in the BEQAM is a violation of the requirements of 10 CFR 50, Appendix B (84-26-04). This problem was previously identified during inspection 50-293/81-02 and a violation was issued at that time.

9. Personnel Radiation Exposure Tracking

The following information is included in this report to assist NRC management personnel in following radiation exposures at the station.

During the month of September, 1984, 374.6 person-rem of radiation exposure was received during outage activities. The September exposure was less than the previous monthly exposure, 440 person-rem. Approximately thirty percent of the September monthly exposure was attributed to the recirculation piping replacement project.

At the end of September, 1984, the total outage exposure since December 10, 1983 was 3,385 person-rem. On October 1, 1984, the projected outage exposure was increased from 3,575 person-rem to 4,025 person-rem. The increase in exposure was attributed to an increase in the scope of outage work and an increase in the projected length of the outage. The latest dose projection assumes that the outage ends on October 31, 1984. The estimate for recirculation replacement project exposures was unchanged in the latest projection, at 1575 person-rem.

10. Management Meetings

During the inspection, licensee management was periodically notified of the preliminary findings by the resident inspectors. A summary was also provided at the conclusion of the inspection and prior to report issuance. No written material concerning the results of this inspection was provided to the licensee during this inspection.