U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	50-293
Report No.:	94-09
Licensee:	Boston Edison Company 800 Boylston Street Boston, Massachusetts 02199
Facility:	Pilgrim Nuclear Power Station
Location:	Plymouth, Massachusetts
Dates:	April 5 - May 9, 1994
Inspectors:	J. Macdonald, Senior Resident Inspector D. Kern, Resident Inspector D. Mannai, Resident Inspector, Susquehanna

Approved by: John B Macdonald - for 6/2/94
R. Conte, Chief, Reactor Projects Section 3A Date

Scope: Resident Inspector safety inspections were conducted in the areas of plant operations, maintenance and surveillance, engineering, and plant support. Initiatives selected for inspection included maintenance of fire barrier integrity, the diesel generator fuel oil sampling program, and the nuclear safety concerns program.

Inspections were performed on backshifts during April 12, 13, 18-21, 26, 28 and May 2, 3, 5, 6, and 9, 1994. Deep back shift inspections were conducted on April 24 (5:00 - 10:00 pm), April 27, 1994 (11:30 am - 6:15 pm), and May 7, 1994 (3:00 - 4:00 pm).

Findings: Performance during this five week period is summarized in the Executive Summary.

Violation: The failure to properly establish fire watches as compensation for degraded fire barriers on several occasions during this inspection period is identified as a violation (50-293/94-09-02).

Unresolved Item: Review of the adequacy of the licensee root cause analysis and subsequent corrective actions to the April 27, 1994 reactor protection and containment isolation systems actuations during filling and venting of the control rod drive system is identified as an unresolved item (50-293/94-09-01).

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EXECUTIVE SUMMARY

Pilgrim Inspection Report 94-09

Plant Operations:

Operations staff maintained proper reactor parameters throughout the extended control rod scram time testing activities. Similar good command and controls were noted during the shutdown and outage period to replace suspect scram solenoid pilot valve diaphragms. Operator response to a control rod that drifted to the full out position during the testing was appropriate. Operators verified proper system status prior to resetting reactor protection and containment isolation systems following actuations during filling and venting of the control rod drive water system.

Maintenance and Surveillance:

Overall, the extended scram time testing activities were effectively controlled. However, on two instances maintenance personnel error resulted in a non-selected rod being individually being scrammed. Operators responded properly to these instances and no adverse consequences resulted. Good control of material traceability was established during the diaphragm replacements. Separately, it appears that an inadequately established system isolation boundary may have contributed to the April 27, 1994 reactor protection and containment isolation system actuations. This issue remains unresolved pending review of final root cause determinations and corrective action identification.

Engineering:

Engineering response to the degraded scram time testing results was excellent. A multi-disciplined issue team was promptly established that evaluated all aspects of the issue. Plant specific test data and recent industry experience were fully reviewed. The determination to shut down following visual inspection of a pre-selected sample of pilot valve diaphragms that exhibited accelerated degradation reflected sound technical judgement and appropriate safety perspectives.

The fuel oil sampling program for the emergency diesel generators was consistent with established standards and was being properly implemented. During the inspection period, several instances were observed in which fire barriers had been degraded without proper compensatory fire watches being established. These instances appear to have identified weaknesses in the implementation of the fire protection program. A Notice of Violation is being issued as a result of these occurrences and the probable associated programmatic weaknesses they represent.

Plant Support:

The radiological survey plan established for the removal of temporary office trailers from the station was observed to have been properly implemented. The recently instituted nuclear safety concerns program has been effectively introduced to the Nuclear Organization. The program was widely previewed to the staff and appears to have developed the attributes of an effective safety concerns initiative.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

At the start of the report period, Pilgrim Nuclear Power Station was operating at approximately 100% of rated power. On April 16, 1994, reactor power was decreased to approximately 50% to perform a thermal backwash of the main condenser and to conduct scram time testing of 17 control rods. Several control rods exhibited slower scram insertion times, one rod exceeded acceptable full insertion time and was declared inoperable, and the test sample population was expanded. Troubleshooting of the inoperable control rod revealed a failed Buna-N exhaust diaphragm and a cracked pressure diaphragm in the scram solenoid pilot valves (SSPVs). Ultimately, the entire core was scram time tested with overall indication of slower but acceptable results. However, visual inspection of the diaphragms in SSPVs of slow and fast control rods indicated various levels of degradation that was not representative of the scram time performance of the associated control rods. Therefore, on April 22, 1994 a reactor shutdown was initiated to replace the SSPV diaphragms on each control rod hydraulic control unit ((HCU), Section 2.2). In addition to the HCU SSPV diaphragm replacements, the 'B' recirculation pump seal was replaced during the shutdown.

On April 27, 1994, with the reactor in cold shutdown and all control rods fully inserted, a reactor protection system and partial primary containment isolation system initiation signal was generated during filling and venting of the control rod drive system (Section 3.2). The systems were reset and preparations for plant startup were continued. On April 28, 1994, at 10:45 pm, the reactor was made critical. The turbine generator was synchronized to the offsite distribution system at 10:44 am, on April 29, 1994. Reactor power was maintained at approximately 25% of rated power to conduct scram time testing of all control rods and to complete augmented offgas system maintenance.

On April 30, 1994, at approximately 11:00 pm, the turbine generator was removed from the offsite distribution system and secured to facilitate the weld repair of a steam extraction line connection. The repairs were completed and the turbine generator was synchronized to the offsite distribution system at 10:26 am, on May 1, 1994. Scram time testing and other power ascension activities were completed satisfactorily, and the reactor achieved 100% rated power at approximately 9:00 pm, on May 2, 1994. The reactor remained at 100% power through the end of the report period.

2.0 PLANT OPERATIONS (71707, 40500, 90701)

2.1 Plant Operations Review

The inspector observed the safe conduct of plant operations (during regular and backshift hours) in the following areas:

Control Room	Fence Line
Reactor Building	(Protected Area)
Diesel Generator Building	Turbine Building
Switchgear Rooms	Intake Structure
Security Facilities	

Control room instruments were independently observed by NRC inspectors and found to be in correlation amongst channels, properly functioning and in conformance with Technical Specifications. Alarms received in the control room were reviewed and discussed with the operators; operators were found cognizant of control board and plant conditions. Control room and shift manning were in accordance with Technical Specification requirements. Posting and control of radiation, high radiation, and contamination areas were appropriate. Workers complied with radiation work permits and appropriately used required personnel monitoring devices.

Plant housekeeping, including the control of flammable and other hazardous materials, was observed. Several instances of uncompensated degraded fire barriers were observed and are documented in Section 4.2 of this report. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout, and lifted lead and jumper logs.

2.2 Control Rod Scram Time Testing

On April 17, 1994, the licensee initiated control rod scram time testing in accordance with procedure 9.9, 'Control Rod Scram Insertion Time Evaluation.' Technical Specification (TS) 4.3.C.2 requires that a minimum of 10% of the control rod drives be scram tested on a rotating basis within each 120 days of operation. Additionally, TS 3.3.C, 'Scram Insertion Times,' Part 1 requires the average scram insertion time of all operable control rods shall be no greater than 0.55 seconds (sec) for the first 10% of insertion from a fully withdrawn initial position. Part 2 of TS 3.3.C, requires the average scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than 0.58 sec for the first 10% of insertion from a fully withdrawn initial position. Parts 1 and 2 also have associated average time limits for 30%, 50%, and 90% control rod insertion. Part 3 of TS 3.3.C, requires the maximum scram insertion time for 90% insertion of any operable control rod to not exceed 7.00 sec.

Initially, 17 of the 145 control rods were selected for testing. Overall, the average 10% insertion times were approximately 0.04 sec slower than the most recent testing. One control rod, CR10-31, that exceeded the 7.00 sec 90% insertion maximum time limitation was de-energized in the fully inserted position and declared inoperable. Troubleshooting of CR 10-31 identified a failed exhaust diaphragm in scram solenoid pilot valve (SSPV), SSPV 118. The pilot valve was repaired and the control rod was scram time tested satisfactorily. However, a second sample of 29 control rods were selected for testing due to the identification of a failed

SSPV diaphragm and the overall slower scram times of the first sample set. Again, the second sample set had approximately 0.02 sec slower insertion times. Additionally, one rod, CR 50-31, had a 10% insertion time of 1.8 sec which was a significant degradation of performance since it had been most recently tested. Troubleshooting of CR 50-31 identified significant degradation of the exhaust diaphragm in the associated SSPV 118. The pilot valve was repaired and the control rod was scram time tested satisfactorily. A third sample set of 28 control rods was selected for testing that again provided slightly slower scram times.

By this point in the testing effort, the licensee had established an issue team with representation from operations, system engineering, and various nuclear engineering disciplines. A temporary modification (TM 94-09) to the instrument air system was installed during testing of the second sample set of control rods that reduced the scram air header pressure from approximately 102 psig to 92 psig. This modification served to reduce the time necessary to vent the header to initiate actuation of the scram inlet and outlet valves. The modification improved test performance slightly. As a result, a decision was made to test the remainder of the core. The testing again identified mixed results with several slow control rods that required the associated pilot valves to be rebuilt. At the conclusion of the testing and maintenance, overall core average scram time testing results met the Technical Specification requirements. However, the reliability of the pilot valve diaphragms and the ability to trend degradation of the diaphragms were in question. Therefore, the licensee established inspection criteria for the visual inspection of the pilot valve diaphragms associated with five control rods with fast scram insertion times with the intention of developing a trendable performance data baseline. However, on April 22, 1994, the visual inspections identified levels of advanced degradation beyond the established criteria and the licensee concluded the appropriate action was to shut down the reactor and replace the diaphragms on the remaining pilot valves.

Subsequently, all 580 SSPV diaphragms (2 diaphragms per SSPV; 2 SSPVs per hydraulic control unit) were replaced during either the week of scram time testing or during the following outage. Scram time testing following reactor restart on April 28, 1994 resulted in a marked overall improvement in core wide average insertion times. Specifically, the core average scram insertion time for all control rods for the first 10% of insertion was 0.49 sec, an improvement of 0.06 sec.

Preliminary licensee causal analysis concluded the slow scram times were the result of premature aging of the diaphragms in the pilot valves. The diaphragms are made of Buna-N, which is the common name for nitrile-butadiene rubber. Previous industry experience established a four year service life for the diaphragms. All 580 SSPV diaphragms were replaced during the 1991 refueling outage and as such the diaphragms remained within the projected service life expectancy. Licensee correlation of the degraded diaphragms to material records indicated the potential for inconsistencies in the manufacturing of the diaphragms that may be specific to certain manufacturing lots. This information was provided to the supplier for further evaluation. Recent similar industry experience would seem to support this initial assessment of inconsistent material control and premature aging. The licensee has provided samples of the degraded diaphragms to the supplier and independently has contracted with its own materials laboratory to establish the material quality of the suspect diaphragms.

Licensee response to this event was appropriate. Upon indication of decreased scram response times, the sample sets were expanded. Additionally, when an apparent material deficiency was identified a conservative action was taken to shutdown and replace all the SSPV diaphragms. The issue team effectively evaluated the applicability of recent industry experience and established a sound inspection criteria for the visual examination of the diaphragms that ultimately led to the decision to shut down. Finally, the data specific to the Pilgrim scram time testing was promptly provided to the involved vendor and entered into the industry experience program. The inspector had no further questions with respect to the licensee response to this event. Additional actions on this matter would be predicated on any potential generic implications that may result from continued assessment of sub-vendor material control of diaphragm manufacturing processes.

2.3 Control Rod Malfunction

On May 1, 1994, at 12:45 a.m., with the reactor at approximately 18 percent power, control rod (CR) 46-43 drifted from position 04 to full out (position 48). At the time of the event, operators were performing scram time testing following replacement of the scram solenoid pilot valve (SSPV) diaphragms (Section 4.1). Following the scram time testing of CR 46-43, operators attempted to withdraw the rod to position 48 to restore it to its original position prior to the test.

During the withdrawal of CR 46-43, the performing operator noted that the rod motion was slower than expected, and released the notch switch and notch override switch. However, the rod did not stop as expected and continued to withdraw. A control rod drift alarm was received. The operator also noted the settle light did not illuminate. The operator entered procedure 2.4.11, 'Control Rod Positioning Malfunctions,' Section 4.2, but rod motion continued. An insert signal had no effect on rod motion. Additionally, the rod worth minimizer (RWM) prevented insertion of the rod since reactor power was below the low power set point (LPSP). The rod continued to withdraw until it reached the full out position. At that time the rod responded to an insert signal. The rod was subsequently disarmed in the fully inserted position and declared inoperable as required by Technical Specifications. Reactor engineering analysis determined no reactor core thermal limits were exceeded. The licensee documented the condition on Problem Report 94-9204 (Level 1 significance).

A Priority 1 Maintenance Request (MR) was initiated to troubleshoot the affected control rod. The licensee determined the control rod drive system was designed such that no single electrical failure would cause a continuous withdraw signal. Initial troubleshooting indicated the control rod motion sequence timer was functional, however some contact chatter was observed. The licensee replaced the sequence timer as a precautionary measure. Instrumentation and control technicians determined high circuit resistance caused the settle light not to illuminate. The relay contacts for the settle light were burnished. System engineering evaluation concluded the most probable cause of the rod withdrawal was sticking of the collet fingers due to impurities in the drive water. A collet finger flush was performed to clear any potential material from the control rod drive. The rod was successfully exercised in the withdraw and insert directions several times satisfactorily. The rod was then declared operable and returned to its required position.

The inspector closely monitored the issue from initial licensee identification of the problem through the subsequent corrective actions. The Inspector expressed concern that initial causal analysis did not consider potential directional control valve malfunction, a scenario described in the Final Safety Analysis Report. Additionally, although a reactor core thermal limits analysis was performed for the specific control rod, a more generic evaluation for other core configurations was not performed. The inspector considered additional evaluation to be appropriate pending a bounding causal analysis determination of the failure mode. The inspector communicated these early assessments to the licensee. As a result, the review of the matter was promptly broadened to include a more comprehensive evaluation of the event and subsequent root cause determination. Additionally, the directional control valves and associated filters were inspected during a planned power reduction. No indications of abnormal material condition were noted. As a precautionary measure, the licensee has added the overhaul of the drive mechanism for CR 46-43 to the outage schedule. The inspector had no additional questions regarding this event.

3.0 MAINTENANCE AND SURVEILLANCE (61726, 62703, 90712)

3.1 Oversight of Control Rod Testing and Corrective Maintenance

During the course of the initial control rod scram time testing and the subsequent post maintenance scram time testing of control rods with newly replaced scram soleniod pilot valve diaphragms, the licensee performed well over 200 individual test evolutions. Overall, control and conduct of the testing and maintenance activities were excellent. Notwithstanding, two exceptions were noted. Specifically, on April 20 and April 22, technicians de-energized a control rod not preselected for scram time testing at that point which caused the unplanned automatic insertion of those individual control rods. In each instance, control room operators properly responded to the unanticipated transient and promptly restored the inserted control rod to its assigned full out position. The second occurrence briefly reduced reactor power below 45 percent, resulting in a more restrictive minimum critical power ratio operating limit (MFLCPR). Four fuel bundles briefly exceeded the MFLCPR limit as a result of the change in flux distribution. Operators raised recirculation flow and promptly returned the reactor to below the MFLCPR limit. The inspector independently reviewed core performance records and verified that operator actions were effective in returning the reactor to below the MFLCPR limit within the time specified by Technical Specifications. The licensee properly addressed each event via the problem report process to identify causal factors and assign corrective actions. The inspector considered these two events to be isolated and not representative of systematic weakness. Maintenance technicians worked closely with system engineers to label and maintain accountability of each SSPV diaphragm that was replaced. This careful control and tracking of SSPV diaphragms was important in supporting root cause analysis efforts as discussed in section 2.2.

The inspector observed that the rebuild and installation of several scram solenoid pilot valves (SSPVs) were accomplished in accordance with station procedure, 3.N 2-16, 'Scram Pilot Maintenance.' The inspector determined that personnel were knowledgeable and followed the procedure. The work packages were thorough and comprehensive. The use of a separate work package, with references, for each hydraulic control unit (HCU) was considered by the inspector to be a maintenance strength. The licensee maintained proper material control during SSPV rebuild. Instrumentation and control technicians were observed to be using properly calibrated torque wrenches. Quality Control (QC) department personnel provided appropriate oversight coverage. Overall, the licensee properly performed the maintenance in accordance with PNPS procedures. One isolated exception observed by the inspector, was the failure to reinstall scram air header cleanliness covers for HCU 22-39. Maintenance technicians promptly installed cleanliness covers when notified of the discrepancy.

3.2 Improper Boundary Isolation during Control Rod Drive System Maintenance

With the reactor shut down, the control rod drive (CRD) system was depressurized and vented to support the replacement of the scram solenoid pilot valve diaphragms. During this time, operators noted that the "B" emergency core cooling system (ECCS) reactor vessel (RV) water level instrument indication was inconsistent with the other RV level indications. This instrument read significantly higher than the other instruments, indicating a the possible presence of air in the instrument reference leg. On April 27, technicians filled and vented the CRD system following completion of maintenance. During this evolution a false low RV water level signal occurred and resulted in several engineered safety feature actuations. The false signal resulted in a full scram signal in addition to reactor building isolation system and Group 2, 3, and 6 primary containment isolation system actuations. Operators promptly verified plant conditions, reset the protective signals, and returned the affected systems to their normal lineups. Following an event critique, operators completed proper vent and refill of the CRD and RV water level reference leg backfill systems.

The licensee conducted an event critique and initiated problem report (PR) 94.9200 to determine the cause of the event and identify corrective actions to preclude recurrence. The critique determined that the RV water level reference leg backfill system (documented in NRC Inspection Report No. 50-293/93-14) had not been isolated prior to CRD system depressurization and remained in service at the time of the fill and vent evolution. Consequently, air and a small pressure pulse were inadvertently applied to the RV level instrument reference legs which caused the 'A' ECCS RV level instrument to momentarily indicate a false low level. Indicated level on this one instrument changed from +31 inches to +7 inches which caused safety systems to actuate as designed. Operators had developed a fill and vent plan in accordance with procedures 2.2.87, 'Control Rod Drive System,' and 2.1.11.1, 'System Fill, Vent, and Drain Instructions.' However, this plan did not direct operators to verify the backfill system to be isolated from the downstream instruments prior to initiating the fill and vent evolution. Preliminary licensee review of the event determined that procedural weaknesses caused the event. Causal analysis and long term corrective actions were in development as part of PR 94.9200 at the close of this report period.

The inspector independently reviewed the event critique and the interim progress of PR 94.9200. The inspector noted that the potential presence of air in the instrument lines was evident prior to the CRD fill and vent evolution. In addition, procedure 2.2.87 specifically directs that the backfill system be isolated when the CRD system is out of service. This was not done. Procedure 2.2.80, 'Reactor Vessel Level, Temperature, and Internal Pressure Instrumentation,' provides specific instruction on how to secure the backfill system and how to place the backfill system in service following CRD maintenance which has the potential to cause air entrainment. This procedure was available, but not used. The issue of procedural adequacy and proper use of procedures remains unresolved pending inspector review of the completed PR 94.9200 and associated corrective actions (UNR 50-293/94-09-01).

4.0 ENGINEERING (71707, 92700, 92701)

4.1 Emergency Diesel Generator Fuel Oil Sampling Program

A reliable supply of high quality fuel oil for the station emergency diesel generators (EDGs) is necessary to support prolonged operation of the EDGs as an emergency power supply in the event that normal power supplies to safety electrical buses is not available. Technical Specifications require that the EDG fuel oil tanks be sampled monthly in accordance with the sampling process described in American Society for Testing and Materials (ASTM) standard D4057-81 or D4177-82. Oil sample analysis must conform to the specifications of ASTM D975-81. The inspector reviewed station procedures, observed fuel oil sampling, interviewed personnel, and reviewed industry analytical standards to determine the adequacy of the implementation of the licensee emergency diesel generator (EDG) fuel oil sampling program established to ensure EDG fuel oil quality.

The EDG fuel oil sampling program is implemented by procedures 7.1.36, 'Diesel Generators' Fuel Oil Sampling and Quality Analysis' and 7.1.55, 'Sampling and Testing of EDG Fuel Oil Deliveries.' The inspector noted that these procedures were detailed and properly encompassed the requirements of ASTM D4057-81 for sample technique and of ASTM D975-81 for analytical sample test method and quality results. The licensee had recently raised the upper limit for the cloud point analysis from 15 degrees Fahrenheit (F) to 25 degrees F. The inspector reviewed the basis for this revision, contained in the response to problem report 93.9428, and concluded that this revision was technically justified since the underground storage tanks are located below the frost line and would not be subject to subfreezing temperatures. The inspector observed sampling of the 'A' and 'B' EDG fuel oil tanks. Technicians were knowledgeable and performed procedure 7.1.36 in a controlled manner. Minor comments regarding the sample equipment and procedure 7.1.36 were discussed with the fuel oil program manager who initiated appropriate actions to further improve the sample process.

In January 1994, laboratory analysis indicated that the 'A' EDG fuel oil was out of specification due to high carbon residue. The licensee declared the EDG inoperable and promptly directed the sample to be reanalyzed. The inspector reviewed the 'A' EDG fuel oil tank sample results for the past two years and the analysis of recent fuel oil deliveries. There was no trend of increasing carbon content and no indication of a potential intrusion of foreign material. Reanalysis of the original 'A' EDG sample was within the specified limit. Backup samples of both the 'A' and 'B' EDG fuel oil tanks were drawn and analyzed to confirm fuel oil quality. These test results indicated normal levels of carbon residue which were well within the specification of ASTM D975-81. Licensee investigation of the one set of anomalous chemical analysis was thorough and concluded that the most likely cause was analysis error at the contracted laboratory. The inspector determined that the licensee EDG fuel oil sampling and inventory control programs were properly implemented to assure a reliable supply of fuel to the EDGs.

4.2 Uncompensated Fire Doors Blocked Open

On May 5, 1994, during a routine tour of the intake structure, the inspector observed two fire doors in the salt service water (SSW) pump rooms that were blocked open without appropriate fire watch compensation. At the time of this observation, the licensee was in the process of overhauling the 'A' SSW pump and had just completed work for the day. The inspector brought the concern to the attention of a security guard, who was present in the area and who unblocked the doors. Additionally, the inspector notified operations personnel in the control room, who generated problem report (PR) 94.9215 to document the concern. Final Safety Analysis Report (FSAR) Section 10.8.4.6, Fire Barrier System, requires that all fire barrier systems providing separation of redundant safe shutdown systems shall be functional at all times when the safe shutdown systems are required to be operable. If one or more fire barrier systems become inoperable, the FSAR requires a continuous fire watch be established on one side of the barrier within one hour. Procedure 8.B.14, 'Fire Protection Limiting Conditions For Operation and Compensatory Measure Fire Watch Requirements,' implements the requirements of the FSAR by establishing the instructions for posting compensatory fire watches.

Fire Door (FD) 243 separates the SSW system from the remainder of the intake structure and is not a door that provides safe shutdown system separation. An hourly fire watch is required as compensation for any degradation of the fire barrier function of FD 243. Fire door (FD) 245 separates the 'A' and 'B' SSW pumps from the 'C' SSW pump and provides safe shutdown system separation as described in the FSAR and requires a continuous fire watch as compensation for any fire barrier degradation. Notwithstanding, both of these doors were blocked open to support various aspects of the pump overhaul without proper fire watch compensation being established.

On May 9, 1994, the licensee conducted a critique of the event. The critique was attended by all involved disciplines. Preliminary facts established at the critique indicated the pump overhaul work plan did not require the doors to be maintained open and therefore fire watch compensation was not requested. However, as the job progressed FD 243 was opened to allow the passage of air hoses. Additionally, FD 245 was blocked open to ease communications with underwater divers located in the vicinity of the SSW pumps. Security aspects of the maintenance activities were properly compensated. The inspector noted the initial licensee conclusion that the involved individuals were not sensitive to the fire protection functions for these doors.

Several other fire protection program deficiencies were identified by the licensee following the inspector observation of the uncompensated intake structure fire doors. Specifically, on May 6, 1994, PR 94.9218 documented three non-FSAR fire doors in the condensate bay area that had tape on the latches, preventing the doors from properly being secured. Additionally, a flatbed truck was parked in the reactor building trucklock May 6-9, 1994, without a proper fire watch being established or a combustible permit being issued (PR 94.9225). A third minor instance was also identified in which a smoke door was blocked open without proper fire watch compensation. These examples would seem to indicate that implementation of the recent organizational restructuring initiative that redistributed fire protection program responsibilities has not been fully effective. The inspector and licensee identified examples of the failure to properly implement the fire protection program as required by FSAR Section 10.8 and station procedure 8.B.14 is a violation (NV 50-293/94-09-02).

5.0 PLANT SUPPORT (71707)

5.1 Radiological Surveys for Release of On-Site Trailers

The licensee relocated several plant organizations to the new engineering support building located outside of the protected area during this past winter. This additional work space eliminated the need to continue the use of temporary trailers for on-site office space. The trailers were located within non-contaminated areas of the site. During the months of March and April 1994 the licensee performed radiological surveys of several trailers that verified no radiological contamination existed and removed the trailers from the site.

The inspector discussed the overall plan to survey and remove the trailers with the Radiological Section Manager. Direct frisk surveys were performed on habitable areas such as floors, doors, walls, ceilings, and roofs. Swipe surveys were performed of the trailer ventilation ducts. The inspector observed technicians performing surveys and reviewed selected survey documentation. The surveys reviewed indicated that the trailers were not contaminated. Surveys were comprehensive and were properly performed. Survey instruments had been calibrated within required periodicities and were correctly used. The inspector concluded that the licensee had established and properly implemented an appropriate survey plan for release of the trailers from the site.

5.2 Nuclear Safety Concerns Program

In January 1994, the licensee issued nuclear organization procedure (NOP) 93A2, 'Nuclear Safety Concerns Program (NSCP),' to formalize a process by which all employees and contractors can report perceived nuclear safety concerns. In conjunction with the issuance of this procedure, station wide training was conducted. A video tape was created in which key managers described the importance of the program and their support for its constructive use by all employees if concerns for nuclear safety are not resolved to the individual's satisfaction using other existing processes. The video was of high quality and clearly conveyed the purpose of the program.

The inspectors reviewed NOP 93A2 and discussed program implementation with the Nuclear Safety Concerns Program Administrator (NSCPA). The Senior Vice President, Nuclear maintains overall responsibility for the implementation, administration, and conduct of the program. Confidentiality and anonymity were mutually recognized as key elements to the success of this program. Procedure 93A2 contains appropriate guidance and process to assure confidentiality and provide periodic feedback to the individual who raises a nuclear safety concern. Although NOP 93A2 does not specify the form in which the feedback will be presented, the NSCPA stated that the intention is for the NSCPA to provide a written final report to the person who initiated the concern. The NSCPA then verbally reviews the report with the individual.

The procedure provides appropriate implementation criteria including time frames and tracking of action item assignments. Confidential records of each nuclear safety concern and the corresponding resolution are maintained for five years. The number of concerns raised through the NSCP during the first four months of 1994 has been relatively small. The majority of nuclear safety concerns identified at Pilgrim Station are addressed through the existing problem report system. Success of the current problem report system is one factor which has limited the number of concerns submitted to the NSCP. Several of the concerns which were initiated via the NSCP did not pertain to nuclear safety and were referred to other existing licensee programs for resolution. The licensee has discussed the NSCP with managers of the primary contracting firms who provide services at Pilgrim Station. Plans to make all contractors aware of the availability of the NSCP are under development. The inspector concluded that the licensee had dedicated appropriate resources and research to the development of the NSCP. Management support was evident and the process of using the nuclear safety concerns program was well publicized. The NSCP, as documented in NOP 93A2, provides an easily accessible process by which individuals can raise nuclear safety concerns which they believe have not been sufficiently resolved using other corrective action processes.

6.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (30702)

6.1 Routine Meetings

At periodic intervals during this inspection, meetings were held with senior BECo plant management to discuss licensee activities and areas of concern to the inspectors. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting on May 27, 1994, summarizing the preliminary findings of this inspection. No proprietary information was identified as being included in the report.

6.2 Other NRC Activities

On April 28, 1994, Mr. Thomas T. Martin, NRC Region I Administrator, conducted a media briefing in Boston, Massachusetts. The Administrator conducts quarterly briefings at different geographical locations throughout the Region to discuss NRC related activities of specific local interest. Topics of discussion included the performance of nuclear power plants in New England and the site decommissioning management program.

On April 11-15, 1994, an NRC Region I radiation protection specialist conducted an inspection of the licensee radiological controls program. Inspection results will be documented in NRC Inspection Report 50-293/94-08.