U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report No. 50-293/87-57

Licensee: Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: December 7, 1987 - January 19, 1988

Inspectors: C. Warren, Senior Resident Inspector
J. Lyash, Resident Inspector
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Approved By: ARB 3-11-88
A. Randy Blough, Chief Date
Reactor Projects Section No. 3B

Areas Inspected: Routine resident inspection of plant operations, radiation protection, physical security, plant events, maintenance, surveillance, outage activities, and reports to the NRC. The inspection consisted of 350 hours of direct inspection. Principal licensee management representatives contacted are listed in Attachment I. Observations made by the NRC Region I, Regional Administrator during a tour on December 8, 1987 are documented in Attachment II of this report. A copy of Attachment II was provided to licensee management for followup.

Results:

Violation: Repeated occurrences of locked high radiation area doors being left open and unattended were identified by the licensee. Problems with high radiation area access control have been previously identified and were the subject of violations during inspections 50-293/87-03 and 50-293/87-11. Corrective actions taken in response to these findings have not prevented their recurrence. (Section 3.b, VIO 87-57-01)

Unresolved Item: The licensee identified that two reactor vessel level gauges were incorrectly installed. A licensee investigation is currently ongoing to determine the cause and to assess the adequacy of post installation test. (Section 4.d, UNR 87-57-02)

Concerns:

1. The licensee experienced safety related equipment malfunctions upon receiving a spurious reactor scram signal on January 17, 1988. (Section 4.d)

2. Inadequate procedures and planning of surveillance tests resulted in unnecessary engineered safety feature actuations. (Section 3.a)

3. Poor preplanning and control of maintenance was noted during an electrical relay replacement. A similar problem was the subject of a violation during inspection 50-293/87-50. (Section 4.c)

4. Weak identification and tracking of lifted leads and jumpers led to a water spill in the high pressure coolant injection system room during the integrated leak rate test. (Section 6.0)

5. The prelube pump for the "B" emergency diesel generator failed to restart during a surveillance test. An identical failure occurred during a loss of offsite power event on November 12, 1987. Licensee followup appeared adequate but the failure root cause has not been identified. (Section 3.b)

6. The inspectors evaluated the erosion of construction dirt into wetlands area. The inspector's independent survey of the area, and the licensee's analyses indicate that the level of activity does not represent a health or safety concern. However, the material should not be allowed to erode. (Section 3.c)

Strengths:

1. The licensee's preparation and execution of the reactor vessel hydrostatic test was well organized and controlled. (Section 5.0)

2. The licensee's response to a January 17, 1988 reactor scram signal and subsequent equipment malfunctions was prompt, thorough and effective. (Section 4.d)

3. Using non-nuclear steam for testing of high pressure coolant injection system and reactor core isolation cooling system enabled the licensee to discover problems which may not have been easily identifiable using nuclear steam due to radiological conditions. (Section 3.b)

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DETAILS

1.0 Summary of Facility Activities

The plant was shutdown on April 12, 1986 for unscheduled maintenance. On July 25, 1986, Boston Edison announced that the outage would be extended to include refueling and completion of certain modifications. The reactor core was defueled on February 13, 1987. The licensee completed fuel reload on October 14, 1987. Reinstallation of the reactor vessel internal components and the vessel head was also subsequently completed.

During this report period, the licensee performed the reactor vessel hydrostatic test and the primary containment integrated leak rate test (ILRT) as described in Sections 5.0 and 6.0. On December 9, 1987, Pilgrim Station conducted a partial participation emergency preparedness exercise. On December 14, 1987 the licensee announced as part of a planned management realignment, the appointment of eight managers to key management positions in the licensee nuclear organization at Pilgrim Station. The details of the management realignment are described in Section 7.0.

NRC inspection activities during the report period included: 1) observation of the licensee's annual emergency preparedness exercise on December 9, 1987, 2) NRC Reactor Operator Licensing examinations were administered to eight candidates on the week of December 7, 1987, 3) observation of the primary containment ILRT and review of the test results during the week of December 21, 1987. The results of these inspections are documented in inspection reports 50-293/87-54, 50-293/87-56, and 50-293/87-58. In addition, representatives of the NRC's Office of Investigation were onsite December 3, December 7, and December 8, 1987 to interview onsite security personnel. On December 8, 1987, the NRC Regional Administrator for Region I, Mr. William T. Russell, toured the plant with the resident inspectors. On January 7, 1988, Dr. Thomas E. Murley, Director of the Office of Nuclear Reactor Regulation (NRR) and other NRC representatives toured the plant with the resident inspectors.

2.0 Followup on Previous Inspection Findings

(Closed) Unresolved Item 82-24-02 - Discrepancies in the Licensee's Response to IE Bulletin 79-08

Previous reviews of this item are documented in the inspection reports 50-293/82-30, 50-293/83-01, 50-293/83-14, and 50-293/84-26. IE Bulletin (IEB) 79-08 and the TMI Action Plan Item II.E.4.2 required licensees to review the containment isolation initiation design and procedures to ensure proper initiation of containment isolation, upon receipt of an automatic containment isolation signal. The licensee provided the results of their review in letters dated April 25, and August 21, 1979.

The licensee stated that the RBCCW supply and return lines, instrument air line, RHR to spent fuel pool cooling tie line, and torus make up line would be manually isolated and that station procedures would specify the requirements for manual isolation if a containment isolation signal was received. This was documented as acceptable by NRC:NRR in letters to the licensee dated December 18, 1979 and April 3, 1980. However, an inspector identified that manual isolation of these lines with qualified valves is not possible. Any valve which is used for primary containment isolation must meet Seismic Class I (FSAR section 12.2) and applicable 10 CFR 50, Appendix J, containment leakage testing criteria. Further, if manual operation of a valve is required to effect containment isolation, the isolation point for the valve must also be accessible under those conditions which make its use necessary.

In response to the inspector's questions, the licensee re-evaluated their response to the IEB 79-08 and TMI Action Plan Item II.E.4.2, and concluded that isolation of these lines is assured by the use of Seismic Class I check valves. The licensee also agreed that isolation for the RBCCW supply line, instrument air line, RHR to spent fuel pool cooling tie line, and torus makeup line cannot be performed by manual valve closure. The RBCCW return line from the drywell can meet the isolation valve criteria with MOV-4002 which is seismic class I, local leak rate tested and can be closed by a control switch located in the main control room. The licensee subsequently submitted a supplemental response to IE Bulletin 79-08 and TMI Action Plan Item II.E.4.2 on October 24, 1984 correcting the previous response. The inspector reviewed the supplemental response and verified that the contents were consistent with the conclusions drawn from the licensee's re-evaluation and the FSAR. Both RBCCW supply line and instrument air line are considered Class C lines in Section 7.3 of the FSAR since they penetrate containment but have no interaction with the primary containment free space or the reactor vessel. According to the original design criteria, a single check valve is provided to attain isolation for a Class C line. These check valves are seismic class I and local leak rate tested. The inspector reviewed the results of local leak rate test data for these check valves which were performed on June 12 and July 26, 1987 and found no discrepancies. The torus makeup line is identified as Class B in Section 7.3 of the FSAR. The torus makeup line is non-essential and ties the condensate transfer system into the RHR test line, which penetrate primary containment and ends below the torus water level. For water-sealed Class B lines such as the torus makeup system, the original plant design bases allow one isolation valve in addition to the water seal to meet isolation requirements. Also, the Safety Evaluation by the NRR on Appendix J Review indicate that Type C testing is not required for valves in lines which terminate below the level of the suppression pool. As for the RHR to spent fuel pool line, the licensee revised the operating procedures 2.2.85, Fuel Pool Cooling and Filtering System, prohibiting the use of the RHR to spent fuel pool lines except in cold shutdown. The inspector had no further questions. This item is closed.

<u>(Closed) Inspector Follow Item (IFI 87-27-02) - Cracking of Surge Ring Brackets in Large GE Motors</u>

On July 2, 1987, IE Information Notice 87-30, Cracking of Surge Ring Brackets in large GE motors, was issued. The purpose of the notice was to alert recipients of a potential for failure of surge ring brackets and cracking of felt blocks in large, vertical electric motors manufactured by General Electric Co. Felt blocks are used in large electric motors to keep the windings separated where they loop back at the end of the stator. The blocks are attached to a surge ring that is held in place by L-shaped surge ring brackets welded to the surge ring and bolted to the motor casing. Failure of these surge ring brackets and cracking of the felt blocks allows movement and wear of the end-turns, leading to a reduction in insulation resistance and possible motor failure. In addition, broken pieces of the surge ring bracket may enter the space between the stator and the rotor, resulting in electrical or mechanical motor degradation.

Following an investigation to determine the applicability of the subject notice to the Pilgrim Station, the licensee found that RHR, core spray, and recirculation pump motors were potentially affected. RHR and core spray pump motors were overhauled on site by GE under contract with the licensee in 1986. The surge ring brackets were not inspected during the overhaul. However, small cracks were found on the "A" and "C" RHR pump motor winding felt blocks. The amount of cracking found was dispositioned by GE to be acceptable and a normal phenomenon found in form-wound motors. On July 27 through August 5, 1987, GE performed a surge ring bracket inspection of the RHR and recirculation pump motors using a boroscope with the motors in place. The inspection of the RHR motors (A thru D) revealed absence of cracks on the surge ring brackets. During the inspection of the "B" recirculation pump motor, it was noted that the recirc motor surge ring bracket construction is of the bolt and stud design, whereas the RHR and core spray motor brackets are of the L-shaped design. The L-shaped design configuration is known to have the potential of cracking, according to the IE Notice 87-30 and the GE letter to the licensee dated July 14, 1987.

During the week of October 26, 1987, "B" core spray pump motor was disassembled and the surge ring brackets inspected by G.E. Due to the geometry of the core spray pump motor internals, there is limited access for the bore scope, therefore, this inspection could not be accomplished without partial disassembly of the rotor. It was verified that the design had 12 brackets per surge ring and two surge rings for the top end turn assembly and two surge rings for the bottom end turn assembly. None of the brackets had indications of cracking. The licensee scheduled the inspection of the "A" core spray pump motor during the next outage because of scheduling conflicts. The licensee indicated that based on the inspection results of the RHR and "B" core spray pump motors, postponement of the "A" core spray pump motor inspection is justified. The licensee also added that the number of operating hours and starts are similar between the A and B core spray pump motors since both core spray systems' testing and surveillance requirements are similar. The inspector had no further questions. This item is closed.

(Closed) Unresolved Item 87-45-05 - Failure to Issue Licensee Event Reports

In inspection report 50-293/87-45 the NRC identified three engineered safety feature actuations which appeared to be reportable under 10 CFR 50.73 but had not been reported by the licensee. The licensee reviewed the three actuations, agreed that they should have been reported and agreed to issue License Event Reports (LER) to document the occurrences. In addition the licensee agreed to perform a review of previous actuations to determine if any additional reports were needed.

During this inspection period the licensee's compliance section conducted a review of all Failure and Malfunction Reports (F&MR) issued from April 1986 through the present. This review identified four F&MRs that fit the description of an ESF actuation under the current BECo interpretation of NUREG 1022. The licensee will submit LERs to document the following ESF actuations at a later date.

-- 4/28/87 Initiation signal to both Emergency Diesel Generators (EDG)

-- 6/7/87 Actuation of Reactor Building Isolation and Standby Gas Treatment System start signal

-- 9/17/87 Auto start of "A" EDG

-- 10/6/87 Reactor Water Cleanup and Shutdown Cooling System Isolation

These LERs will be reviewed upon issue as part of the normal resident inspection program. The inspector has reviewed the licensee's actions in addressing open item 87-45-05 and is satisfied that those actions were thorough and timely. This item is closed.

3.0 Routine Periodic Inspections

The inspectors routinely toured the facility during normal and backshift hours to assess general plant and equipment conditions, housekeeping, and adherence to fire protection, security and radiological control measures. Inspections were conducted between 10:00 p.m. and 6:00 a.m. on January 17, 18, and 19, 1988 for a total of four hours and during the weekends of December 12, 19, 27, 1987 and January 3, 9, 17, 1988 for a total of 17 hours. Ongoing work activities were monitored to verify that they were being conducted in accordance with approved administrative and technical procedures, and that proper communications with the control room staff had been established. The inspector observed valve, instrument and electrical equipment lineups in the field to ensure that they were consistent with system operability requirements and operating procedures.

During tours of the control room the inspectors verified proper staffing, access control and operator attentiveness. Adherence to procedures and limiting conditions for operations was evaluated. The inspectors examined equipment lineup and operability, instrument traces and status of control room annunciators. Various control room logs and other available licensee documentation were reviewed.

The inspector observed and reviewed outage, maintenance and problem investigation activities to verify compliance with regulations, procedures, codes and standards. Involvement of QA/QC, safety tag use, personnel qualifications, fire protection precautions, retest requirements, and reportability were assessed.

The inspector observed tests to verify performance in accordance with approved procedures and LCO's, collection of valid test results, removal and restoration of equipment, and deficiency review and resolution.

Radiological controls were observed on a routine basis during the reporting period. Standard industry radiological work practices, conformance to radiological control procedures and 10 CFR Part 20 requirements were observed. Independent surveys of radiological boundaries and random surveys of nonradiological points throughout the facility were taken by the inspector.

Checks were made to determine whether security conditions met regulatory requirements, the physical security plan, and approved procedures. Those checks included security staffing, protected and vital area barriers, personnel identification, access control, badging, and compensatory measures when required.

a. Surveillance Testing

-- Diesel Generator Prelube Pump Failure

On December 13, 1987, the prelube pump for the "B" emergency diesel generator (EDG) failed to restart on demand during a routine surveillance test. Upon disassembly it was identified that a small piece of metal had become lodged between the pump rotor and idler gear. The interference from the metal caused the pump motor breaker to trip on pump start. An identical failure occurred during a loss of offsite power event on November 12, 1987. In that case the failure caused a lengthy delay in returning an idle diesel to service. While not required for diesel operation, the prelube system reduces EDG bearing wear during equipment start.

In response to the failures, the licensee drained and inspected the lube oil sump, and disassembled and inspected the lube oil filters, strainers and heater. The lube oil heater was found to have failed in the energized mode resulting in significant carbon deposits in the heater and filter. No appreciable deposits were found in the lube oil sump. In addition, a piece of filter element packaging material was found in the lube oil filter housing. No foreign material which could have contributed to the prelube pump failure, however, was found. The pump was replaced and the diesel was returned to service. No additional failures occurred during the inspection period. The two pumps which failed had in-sequence serial numbers. Licensee Quality Control personnel performed magnetic particle and dye-penetrant testing of the internals of a third in-sequence pump in the warehouse. No flaws were noted. The licensee is pursuing the root cause of the failures in cooperation with the pump vendor, Viking Pump. The licensee stated at the exit interview that the "A" EDG prelube pump and lube oil heater would be inspected during the next "A" diesel outage. The inspector will continue to monitor licensee followup to this problem.

- <u>Steam Testing of the High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems</u>

The licensee completed full pressure steam testing of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) system turbines by utilizing temporary oil fired auxiliary boilers as a source of non-nuclear steam. The full pressure steam testing is part of a post-maintenance and system operability check. Both HPCI and RCIC systems were overhauled during the current outage. Utilizing temporary test procedures TP 87-198 and TP 87-199, the HPCI/RCIC testing included turbine overspeed trip, pump full flow capacity and operation from the alternate shutdown panels. Also during the test, the suction path was changed from the condensate storage tank to the torus and back.

During the testing, several problems were identified by the licensee in both HPCI and RCIC systems. In HPCI, problems with the governor control system were noted including a minor oil leak in the servo-motor. Steam leaks at gauges and turbine drain line were also discovered. In RCIC, the licensee discovered a previously installed blank flange in the turbine steam leak off line which caused steam leaks. A few problems were also noted on the RCIC governor control system. The licensee is in the process of dispositioning these items. The inspector noted that using non-nuclear steam for the testing enabled the licensee to discover problems which may not have been easily identifiable using nuclear steam due to the radiological conditions. The inspector will review the results of the tests and dispositioning of the problems identified during the tests.

- Incorrect Installation of Fire Dampers

On December 17, 1987, during performance of a routine surveillance test the licensee inadvertently actuated two fire dampers. One of the dampers failed to fully close due to interference with a hook used to secure it in the open position. When the fusible link was energized, the metal damper retaining strap should have fallen away allowing full closure. The hook attaching the strap to the fusible link was oriented with the open side toward the damper. The damper caught on the hook and remained partially open. Upon discovery the licensee immediately stationed fire watches at all areas containing suspect dampers. Inspections were promptly conducted and it was identified that all of the installed hooks were oriented in this manner. The hooks were repositioned so that the open side faces away from the damper. Three dampers were inaccessible and compensatory measures remain in place pending inspection.

The dampers were originally supplied to the licensee without the hooks. A revision to the plant design change (PDC) package added the hooks to facilitate surveillance testing. Installation instructions contained in the PDC specified hook orientation with the open side toward the damper. The vendor data sheet supplied by Air Balance Inc. also showed the hook installed in this manner.

Licensee event report (LER) 87-020-00 was issued describing the problem and corrective actions taken. The LER states that preliminary licensee assessment of the issue determined that it did not meet the reporting threshold of 10 CFR Part 21. The inspector discussed the Part 21 reportability with the licensee's Nuclear Engineering Department (NED). NED personnel stated that the failure mechanism was created by the licensee when the hook was added. In addition the presence of mitigating factors such as fire detection and suppression, and control of combustible materials support the conclusion that a substantial safety hazard did not exist. The licensee also feels that LER 87-020-00 contains sufficient information to clearly define the problem. The inspector had no further questions in this area.

The inspector examined two dampers in the cable spreading room to verify that the hooks had been reoriented. Both hooks had been modified, however, neither of the dampers had locking rings installed at the hook to retaining strap connection as required by the installation instructions in the PDC. The licensee reviewed the function of the locking rings and concluded that they were not required. A change to the PDC was initiated to delete the ring. The inspector had no further questions.

b. Radiation Protection and Chemistry

- Locked High Radiation Area Access Control

During the period covered by inspection report 87-57, four instances occurred in which the licensee failed to properly control access to areas that had been designated as locked high radiation areas. In three of these cases, doors to locked high radiation areas were found closed but not locked and in the fourth case a door into a locked high radiation area was found to not be on the list of doors that were being controlled under the locked high radiation area door procedure.

On December 15, 1987, a contract painter failed to check that the door to the locked high radiation area he was exiting was properly latched. The unlatched door was identified during the next routine check of high radiation area doors. Licensee personnel immediately latched the door and initiated a radiological occurrence report (ROR) to document the occurrence and track all actions taken during the investigation. Surveys of the area showed no dose rates greater than 1000 millirems per hour (MR/hr). Interviews with the individual involved determined that the procedures and requirements were well understood and that the HP technician had informed them of their responsibilities prior to entry into the area.

On December 27, 1987, and again on January 8, 1988, instances similar to the one described above took place. In both cases the licensee initiated RORs and took steps to determine: 1) who had been in the area, 2) were they aware of the procedure, and 3) had they been properly briefed prior to entry into the areas involved. In both of these cases the root cause has been determined as personnel error.

In one instance the licensee identified that one of the multiple doors into an area classified as a locked high radiation area was not on the list of doors to be checked on a routine basis. The door was immediately checked and found to be locked. Records have been audited to determine if any unauthorized entry into the area had occurred and no instances were identified. The door has been placed on the list and is now routinely checked.

The inspector reviewed licensee actions as a result of these instances and is satisfied that in all cases, the immediate and followup actions were timely and complete. Surveys taken were comprehensive and conducted almost immediately after discovery of unlocked areas. Dose calculations were performed and dosimetry read in all cases. Involvement by senior HP and plant management was evident in all instances.

Inadequate control of locked high radiation areas has been an area of longstanding NRC concern. Notices of Violation have been issued in the past, during inspections 50-293/87-03, 50-293/87-11, and 50-293/87-19 which addressed these concerns. In regard to these violations the licensee instituted corrective actions which have been successful in addressing segments of the problem but have not been successful in preventing recurrence of events involving high radiation area door control.

The inspector has independently reviewed the licensee's program for control of high radiation areas and high radiation area key control and has found them adequate. Although the programs themselves are adequate and personnel have been trained on those programs, instances still occur where locked high radiation areas are not adequately controlled.

Based on review of these four instances coupled with the review of Unresolved Item 87-50-08, the inspector determined that the licensee actions in response to these previous findings have not prevented recurrence. Failure to comply with the requirements of Technical Specification 6.11 and Implementing Procedure 6.1-012 is an apparent violation of NRC requirements as documented in Appendix A of the cover letter to this report (87-57-01). Licensee response to Appendix A should include those measures taken to insure that corrective actions are effective and lasting.

- <u>Contaminated Clothing Offsite</u>

On December 17, 1987, at 7:26 p.m. hours a Bechtel pipefitter who was exiting the reactor building, set off a whole body portal monitor alarm. The portal monitor indicated contamination of his chest area and left hand. The health physics technician on duty at the access point removed the individual from the portal monitor and began performing a survey using a RM-14 with DT 260 probe. The HP technician identified; 1) contamination on the individual's left hand, 1-2 thousand dpm per 100 square centimeters (K DPM), which was removed by washing, 2) contamination on the shirt in both the chest (80K DPM) and lower stomach area (1K DPM). The shirt contamination was removed by tape (80K DPM) and washing with soap and water (1K DPM). The employee, now wearing an undershirt and trousers, was then sent to clear the portal monitor which again alarmed and indicated contamination in the chest area. The HP technician again surveyed the individual and identified contamination on the undershirt in the chest area (70K DPM). The individual was then sent into the portal monitor bare chested and was cleared. The individual was given his outer shirt, which was still wet from decontamination and cleared through portal monitor. At this point, the individual removed the wet shirt, put on his jacket, cleared the portal monitor again, and left for his home.

Upon returning to work December 18, 1987, the individual was given a whole body count to determine if any internal contamination had occurred. The whole body count showed no internal contamination. After completion of the whole body count the individual was interviewed to determine how he had been contaminated, where the occurrence took place and how long he was contaminated prior to detection, to calculate skin dose received.

The interview revealed that the individual had been contaminated when he disconnected a partially pressurized service air hose and depressurized it. The interview also revealed that the individual used the portal monitor at the 91 ft. elevation of the reactor building, received an alarm, did not call for HP assistance but instead tried to decontaminate himself prior to proceeding to the reactor building access. Station procedures require that an individual who finds himself contaminated is to call health physics for assistance. The individual stated that he was aware of this requirement. During the interview the individual expressed concern about whether his heavy winter jacket could have shielded the contamination on his shirt and undershirt from detection by the portal monitors. To demonstrate that this could not happen, a HP supervisor placed plastic bags, which contained the contamination removed from his shirt, inside the coat and attempted to exit through two portals. The portal monitors alarmed on each attempt. The individual appeared satisfied with the demonstration put his jacket back on, with the plastic bags removed and attempted to leave the reactor building. An alarm was actuated on the portal monitor and contamination was indicated on the left arm. The on duty HP technician removed the individual from the portal monitor and identified 3K DPM contamination on the upper right sleeve (outside) of the jacket even though the jacket had not been worn into the reactor building. At this juncture the individual expressed concern over whether the shirt that he had worn the previous day could still be contaminated. The licensee had a HP technician accompany the individual to his home. The individual's shirt was found to be contaminated, was bagged and returned to the site. Surveys of the individual's home and vehicle identified no further contamination.

Efforts to determine how the contaminated shirt was worn through the portal monitors without setting of an alarm yielded positive results. The individual stated that he had purposely kept himself away from the portal monitor in an attempt to keep his wet shirt away from his skin. The licensee taped the plastic bags, with the contamination in them, back onto the shirt and an HP supervisor attempted to pass through the portal monitors by mimicking the body posture used by the individual when he cleared the monitor. The HP supervisor was able to pass through six different monitors without setting off an alarm. The HP supervisor then used the portal monitors in the correct manner and all six monitors alarmed proving that the equipment was functional.

The licensee has evaluated the occurrence to identify the root causes and immediately implemented corrective action. This occurrence was caused by one sequence of events that involved two distinct personnel errors. The primary cause involved the failure of the HP technician to perform an adequate survey of the contaminated individual's clothing when the portal monitor alarm was received. The second problem involved the failure to properly use the installed portal monitors at the reactor building access.

In addition to personnel interviews to identify the sequence of events the licensee also reviewed procedural adequacy, personnel training and portal monitor calibration and performance. These reviews verified that training was adequate and portal monitor performance was as designed. Procedures for control of contaminated individuals at the reactor building access did not specifically require that all articles of clothing require a 100% frisk prior to this occurrence. Instructions have been posted at the reactor building access which now clarify the procedure to be followed when an individual is found to be contaminated.

The portal monitors in use at Pilgrim do not presently have a switch at chest level which must be actuated to start the monitoring process. Lack of this feature allowed the individual wearing a contaminated shirt to lean away from the machine sufficiently to clear the monitor without any alarm. The licensee has determined that the manufacture of the portal monitor now produces a chest high switch for the installed model and will install them in the future.

Calculations have been performed by the licensee to determine the radiation dose received by the individual and the amount of radioactive material that was released from the site on the contaminated shirt. The results of these calculations show that the individual received a localized radiation dose to the skin of 260 MRem, which is below the federal limits for skin exposure, and that the amount of radioactive material on the individuals clothing was 0.2 microcuries which meets the federal criteria as an exempt quantity of Co-60. The inspector is satisfied with the licensee's analysis and corrective actions and has no further questions.

- Allegation of Improper Disposal of Radioactively Contaminated Shrubs (RI-87-A-0107)

On August 31 and September 11, 1987, the NRC resident office at Pilgrim received allegations that radioactively contaminated shrubs had been removed from the site and improperly disposed. The alleged improper disposal occurred on July 23, August 26 and August 28, 1987. During this time period the licensee removed a large number of shrubs from various areas of the site, including those planted near the old administration building and the switchyard. The shrubs were removed to facilitate site construction activities and to alleviate certain security concerns. Upon receipt of the first allegation on August 31, 1987 the NRC requested that the licensee perform an evaluation and provide the results for review. In addition an independent NRC review was initiated.

Resident and specialist inspectors reviewed the licensee's conclusions. The licensee evaluated material release records and interviewed personnel regarding removal of shrubs during the week of July 20, 1987. Several truckloads of shrubs that were transported offsite during the midnight shift on July 24 were examined in detail. Because trace amounts of Cobalt-60 had previously been found in soil onsite, some of the shrubs had the soil removed from the roots prior to release. Each shrub was hand surveyed and found to meet established offsite release criteria. They were transported first to the licensee's shorefront area and later to a dump site on licensee property. The licensee concluded that the shrubs had been adequately surveyed and that no radioactive material had been improperly released.

The resident inspectors reviewed the licensee's program for control of release of material from the site. This area was also evaluated by NRC specialist inspectors during inspection 50-293/87-19. Both inspections concluded that appropriate surveys and release limits have been established and implemented. Resident and specialist inspectors examined licensee release records for the dates in question to verify that vehicles leaving the protected area had been properly surveyed. No discrepancies were identified. An NRC resident inspector accompanied by a licensee representative collected four samples of the shrubs which had been deposited in the dump site discussed above. Each of the four samples consisted of root, branch and foliage clippings from a number of different shrubs. The samples were independently analyzed by the NRC. Three of the samples indicated no contamination. One sample indicated only trace levels of Cobalt -60. Measurements showed that the amount of CO-60 present in this sample was about 2% of the average radioactivity typically found in soil due to naturally occurring isotopes.

The licensee's program for release of material from the site appears adequate. Appropriate survey techniques and release limits have been established. Review of records confirmed that the program is being implemented. Samples of the shrubs collected by the NRC showed zero or negligible contamination and pose no health and safety concern. Based on the above this allegation is considered closed. NRC Region I staff provided status briefings concerning this allegation to Senator Kennedy's staff and to the Massachusetts Department of Public Health.

- Allegation of Airborne Radioactivity in the Trash Compaction Facility (RI-87-A-0120)

On October 5, 1987, the resident office received an anonymous allegation that personnel working at the sort table in the trash compaction facility (TCF) were being routinely exposed to airborne radioactive contamination. The alleger stated that the two filter systems designed to treat exhaust air from the sort table prior to discharge into the room were not functioning, and that the filter differential pressure alarm circuits had been disabled.

On October 7 and 8, 1987, NRC specialist inspectors toured the TCF and examined the design and condition of the equipment. The sort table is used to separate contaminated materials for compaction and disposal. Potentially contaminated air is exhausted from the table, passed through two filters operating in parallel, and released into the room. Airborne radiation levels in the room are measured by means of a separate air monitor which is operated whenever the sorting table is used. The alarm is typically set at 3 X 10 -10 (3E-10) microcuries per cubic centimeter (cc). In addition the filters are surveyed daily and changed if contact dose rates exceed 2mR per hour. The inspectors also examined the trash compaction unit in the area and found that similar controls had been applied. Based on the above, no immediate health and safety concerns were indicated.

On January 15, 1988, the resident inspectors toured the TCF, examined equipment operation and interviewed licensee and contractor personnel involved in ongoing work activities. A radiation work permit specifying protective clothing, health physics coverage, and use of a continuous air monitor is in place to control work at the sort table. Personnel involved stated that trash bags were surveyed prior to sorting and rejected if radiation levels exceeded 5mr/hr, if they contained liquid, or if any powdery material was present. The health physics technician on duty stated that filter radiation levels are monitored daily.

Workers and health physics personnel also stated that filter differential pressure (dp) instruments are monitored to detect filter plugging, however no one had been clearly assigned this responsibility and no dp limit was established. The inspector observed the operation of the table and noted that the "filter restricted" alarm actuated for one of the two filters. The alarm actuated for the filter displaying the lower differential pressure. When questioned workers stated that much of the monitoring and alarm circuitry for the table was not functional, and that the filter alarm was not reliable. The table was originally part of a larger processing system and much of the disconnected circuitry was intended to perform functions which are no longer needed. The inspector verified that current filter dp readings are consistent with the manufactures name plate data.

It appears that the general process applied, including inspection and survey of trash bags prior to sorting, daily filter surveys and continuous air monitoring would preclude airborne radioactivity problems. Based on the above this allegation is closed. However, the inspector noted that no work instructions existed describing the controls applied and equipment monitoring requirements. When discussed with licensee radiation protection management they promptly committed to review the situation and issue appropriate guidance. This was confirmed during the inspector's exit interview.

- Erosion of Construction Dirt into Wetland

On January 15, 1988, at 5:45 p.m. the licensee made an ENS notification in accordance with 10 CFR 50.72 (b)(2)(vi) which requires the licensee to inform the NRC of an event or situation related to health and safety of public for which a news release was made or notification of another government agency has been made. During routine environmental monitoring, the licensee observed erosion from a pile of construction dirt into an adjacent licensee controlled wetland. The Plymouth Conservation Commission and the Massachusetts Department of Public Health were notified and the press release was made by the licensee. Also on January 16, 1988 two representatives from the Plymouth Conservation Commission toured the area.

In the last several years during onsite excavation for plant modifications, dirt, asphalt and concrete containing low levels of contamination were stored in a fenced in storage area outside the protected area on the licensee's property. The licensee estimated that the storage area contains 110,000 cubic feet of material. Before removal from the protected area, samples of material were obtained and isotopic analyses was performed by the licensee. The activity found was reasonably uniform at levels of 10(1E-6) and 10(1E-7) microcuries of Cobalt-60 and Cesium-137 per gram. Sampling and storage of this material was previously reviewed during inspection 50-293/87-18. On January 21, 1988 the inspector toured the area, accompanied by a licensee health physics technician, and performed a survey of the storage area and found no detectable radiation above background levels. During the tour the inspector noted that bales of hay had been put around the perimeter of the fence which borders wetlands area to prevent further erosion of material. The fenced in storage area was secured with a locked gate. The inspector's survey of the area and review of licensee's analyses indicate that the level of activity does not represent a health or safety concern. However, the inspector raised a concern to the licensee management that the material should not be allowed to erode. The inspectors will continue to monitor the licensee actions in formulating long term solution to properly dispose of the material.

n. Fire Protection

On January 17, 1988, at 4:55 a.m. the control room received a report from a security guard of smoke coming from a contractor lavatory trailer, which is located adjacent to the Bechtel warehouse inside the protected area fence. The onshift fire brigade chief was dispatched to the scene and confirmed smoke and smoldering in the area. The fire brigade was immediately dispatched and fire was extinguished using a portable dry chemical extinguisher and a hose from a nearby hydrant house. Electrical maintenance was called to shut off the power to the trailer. By 5:30 a.m., the fire brigade members had cleared the scene and a continuous fire watch was posted in the area. The cause of the fire was believed to be overheating of an overhead heating unit for the trailer. No personnel injury occurred. The inspector toured the scene with a licensee fire protection engineer on January 18, 1988. Minor damage to a small area of the ceiling in the trailer was observed. The Plymouth Fire Department was notified by the licensee in the morning of January 18, 1988.

4.0 Review of Plant Events

The inspectors followed up on events occurring during the period to determine if licensee response was thorough and effective. Independent reviews of the events were conducted to verify the accuracy and completeness of licensee information.

a. <u>Spurious Isolations of RHR Shutdown Cooling System</u>

On December 7, 1987, at 2:28 p.m., an inadvertent isolation of both inboard and outboard containment isolation valves on the RHR shutdown cooling suction line occurred. Preparation for the reactor vessel hydrostatic test was in progress. As part of the hydrostatic test procedure, a technician was installing an electrical jumper in the primary containment isolation system logic panel C-941 to bypass the reactor coolant system (RCS) high pressure interlock on the inboard isolation valve. When the termination screws were loosened to install the jumper, the leads lost contact and caused a false high pressure isolation signal. RHR was in its shutdown cooling mode when the isolation signal was generated, and the shutdown cooling suction valves (MOV 1001-47, 1000-50) automatically closed as designed. Coincident with the closure of the valves, the "A" and "C" RHR pumps tripped automatically to protect the pumps from loss of adequate suction. The licensee determined the actuation was due to a personnel error. The licensee revised Procedure 2.1.8.1, Class I System Hydrostatic Test, to caution the I&C technician of potential isolation of RHR shutdown cooling system while installing the jumper.

On December 8, 1987, at 9:45 p.m., the inboard isolation valve (MOV 1001-50) on the RHR shutdown cooling suction line automatically closed. The automatic isolation occurred when the plant reached 100 psig during pressurization for performance of the class I hydrostatic test. The outboard isolation valve (MOV 1001-47) was already closed to form a pressure boundary for the test. The licensee's investigation determined that the cause of the isolation was that Procedure 2.1.8.1 did not identify all the jumpers necessary to bypass the RCS high pressure interlock on the inboard isolation valve.

As immediate corrective action, the licensee halted the pressurization of RCS and reviewed the logic prints. The licensee revised Procedure 2.1.8.1 to reflect the need to install an additional jumper in panel C-942. In reviewing this event along with other similar events documented in previous inspection reports, the inspector noted that inadequate planning and inadequate procedures appear to be a common root cause for several ESF actuations which occurred on September 17, September 22, October 15 and October 24, 1987. The inspector expressed this concern at the exit meeting with licensee management. The licensee informed the inspector that the Technical Group is in the process of developing generic guidance for isolating or jumpering an electrical component which may cause inadvertent safety system actuations. The inspector will continue to monitor the effectiveness of licensee's corrective action to prevent further ESF actuations due to inadequate planning and inadequate procedures.

b. Reactor Water Cleanup System Spurious Isolation

On December 17, 1987, at 11:05 a.m., the inboard primary containment isolation valve on the reactor water cleanup (RWCU) system suction line automatically isolated. I&C technicians conducting a routine surveillance of the RWCU high area temperature isolation logic inadvertently grounded a lead which had been lifted during the test. Grounding the lead resulted in a blown logic power fuse and isolation of the valve (MOV 1201-2). Following investigation by the control room supervisor, the fuse was replaced and the isolation was reset. The licensee's investigation concluded that the root cause is a personnel error. The licensee informed the inspector that the procedure, 8.M.2-1.2.2, Reactor Water Cleanup Area High Temperature, will be revised to provide cautions to the control room operators and the I&C technicians. Also, an effort is ongoing to review recent ESF actuations caused by personnel error to formulate appropriate corrective actions.

c. Engineered Safety Feature Actuations Due to a Failed Logic Relay

On January 6, 1988, at 2:50 p.m., the coil of primary containment isolation system (PCIS) electrical relay 16A-K57 failed, creating a fault and resulting in blown logic power fuses. The deenergization of this portion of the PCIS logic caused a partial primary containment isolation along with a reactor building isolation and start of the "B" Standby Gas Treatment system (SBGT). The licensee notified the NRC at 5:12 p.m. via ENS. The failed relay was a GE type CR120A relay. The licensee has experienced several failures of this type of relay in the last few years. The licensee's evaluation of this high failure rate and corrective actions to address it are described in the inspection report 50-293/87-50.

On January 7, 1988, the inspector reviewed maintenance request (MR) 88-9 which had been initiated to investigate the cause of the above mentioned ESF actuations and to replace the blown fuse and the faulty relay. The inspector noted that the relay replacement was performed using only procedure 3.M.1-11, Routine Maintenance. This procedure contains general guidance and its stated use is for performing maintenance activities which are not complicated or critical enough to require detailed written procedures. In this case, no step-by-step instruction was initiated to control the sequence of work, to control and tag lifted leads and jumpers, and to ensure verification and independent verification of system restoration. A similar problem involving lack of a sufficiently detailed controlling procedure and the appropriate reviews during an electric relay replacement on November 24, 1987 was the subject of a violation as documented in the inspection report 50-293/87-50. The licensee informed the inspector that the corrective actions to address the violation are being formulated and will be submitted to the NRC.

d. Spurious Reactor Protection System Actuation

On January 17, 1988, at 1:13 a.m., a spurious reactor scram signal was generated during the performance of a reactor level instrument calibration. The full scram signal on low water level was received due to a disturbance in the reactor water instrument line when an I&C technician was valving a level instrument (LI-263-59A) back in service. The Rosemount level transmitters (LT-263-57 A&B) which initiated the scram signal are on the same instrument rack. The licensee's preliminary investigation indicated that the root cause of the event is attributed to a combination of personnel error and inadequate procedure. The investigation also identified that the level instruments (LI-263-59 A&B) were incorrectly installed in that the sensing lines were reversed. The new Barton level instruments (LI-263-59 A&B) were recently installed during this outage and would only be used for local indication during a shutdown from outside the control room. The licensee is currently reviewing the plant design change (PDC 85-07) records and post-installation test data to determine the cause. Surveillance test records are also being reviewed by the licensee. This item is unresolved pending the completion of the licensee investigation (87-57-02).

Upon receiving the spurious scram the control room staff noted that scram discharge instrument volume (SDIV) vent valve CV302-23B primary containment vent and purge valves A05044B and A05035B and one of two redundant secondary containment isolation dampers in each line did not close. In addition the "B" standby gas treatment system (SGTS) did not start. Based on the initiating event, these components should have actuated. The licensee notified the NRC of the failures via ENS at 5:00 a.m. on January 17, 1988.

The control room staff conducted an immediate critique with available I&C personnel, and documented observations for management followup. Later on January 17, the licensee inspected the physical condition of the SDIV vent and drain valves and noted paint on the stem of CV302-23B. The paint was removed and the valve successfully stroke timed. The licensee held a second critique with management representatives on the morning of January 18, 1988 to assess the situation. Subsequently, a walkdown of involved isolation logic components was performed to verify relay contact configuration and to identify any jumpers or lifted leads. This walkdown was performed to the extent possible without disturbing components. No discrepancies were noted. Early on January 19, the licensee performed a test in which a reactor scram was intentionally initiated. The same equipment failed to actuate as during the January 17 scram. Based on this licensee management stopped all work on the affected components. A task force composed of members from the technical staff, systems group, I&C and operations was designated to investigate the incident. This team reviewed available information and developed an action plan.

Walkdowns of the air system piping and components supplying motive air to SDIV vent valve CV302-23B were performed to verify that the as built configuration is in accordance with design doc[illegible]s and that components are in good physical condition. No discrepancies were identified. Valves CV302-23B and CV302-22B are supplied air by the same solenoid operated valves. The licensee deenergized these solenoid valves and observed that CV302-22B closed while CV302-23B did not. This indicates a mechanical problem with the valve or operator. The licensee was identifying replacement parts and preparing to disassemble the valve by the close of the inspection period. The inspectors will continue to monitor licensee followup to this failure.

Licensee review of logic drawings confirmed that the remaining equipment which had not properly actuated shared common isolation logic components. A series of surveillance tests was performed to allow monitoring of key relay actuations. A single contact on a General Electric (GE) HFA relay was determined to be misfunctioning. The contact is required to close when an isolation signal is received, actuating the affected equipment. However, contact resistance remained high with the contact closed. The relay was replaced and the system successfully tested. The licensee contacted GE to coordinate disassembly and inspection of the relay. Dissassembly had not begun by the close of the inspection period. The inspector will continue to monitor licensee investigation of this failure.

The inspector expressed concern that three separate equipment malfunctions had occurred during the inadvertent actuation. This may reflect weakness in the surveillance and post-work test program. However, the licensee's response to the actuation and subsequent malfunctions was prompt, thorough and effec[illegible]ve. Control room operators quickly recognized each of the failures. They held a critique on the same shift with involved personnel. Critique observations were clearly documented and provided to management. An additional critique with management present established priorities. Action was taken to freeze equipment until an investigation plan could be developed and implemented. Followup was well coordinated and involved representatives of several portions of the organization. In this case licensee commitment to determining and correcting the problem root cause was evident.

Review of Reactor Vessel Hydrostatic Test Procedure and Test Results

During [illegible] inspection period the licensee completed the reactor vessel [illegible] est. Several reactor vessel instrument nozzles were repaired [illegible] outage, prompting performance of a hydrostatic test rather [illegible] em leakage test. The reactor vessel reached minimum test [illegible] all inspections were completed on December 9, 1987. Only [illegible] age associated with mechanical connections, such as flanges and [illegible] ing was identified. The reactor vessel was depressurized on Dece[illegible] 12, 1987 after completion of excess flow check valve testing.

The inspector reviewed the licensee's hydrostatic test procedure to verify that appropriate prerequisites, precautions and instructions had been included. A sample of valve lineups was reviewed to determine the adequacy of established test boundaries. Completed valve lineups were also examined. Control of temporary electrical and mechanical jumpers was evaluated to ensure proper documentation and restoration. The inspector observed installed pressure instrumentation and verified appropriate range and calibration status. The adequacy of staffing to support test performance was periodically verified. The inspector reviewed test results and discussed them with engineering, operations, and quality control personnel to ensure that test changes were properly processed, adequate inspections were conducted, and that inspection results were promptly dispositioned.

The licensee's preparation for and execution of the test was generally well organized and controlled. Procedures for test performance and conduct of visual inspections were clear and comprehensive. A detailed Quality Control (QC) work instruction was developed specifying components and piping requiring inspection. Inspection assignments were broken down by location, elevation and component. This QC instruction also included a series of piping diagrams depicting the test boundaries which were utilized to assist in inspection performance and documentation. The licensee's Technical Engineering Section, Quality Control staff and Nuclear Engineering Department each reviewed test boundary adequacy. Inspection results were well documented, and maintenance requests were promptly initiated to correct identified leakage.

The licensee experienced two shutdown cooling isolations during implementation of the test procedure. These isolations are discussed in detail in section 4.a of this report. During the test the licensee identified leakage past the seal ring at the stuffing box to pump casing joint on both recirculation pumps. Leakage flow was estimated to be one to two gallons per minute for each pump. The leakage wet the pump casings and portions of the suction piping, and acceptable inspections could not be completed in these areas. The licensee stated that similar leakage on at least one of the pumps was noted during the last outage. That leak sealed as system temperature increased during startup. The licensee believes that the leakage observed during the recent test will also stop as temperature is increased, and no pump repairs are planned. The licensee stated at the inspector's exit interview that the pump casings and suction piping will be reinspected during startup.

The inspector noted that the test procedure did not contain valve lineups for manual instrument isolation valves within the test boundary. Many instruments and a significant portion of instrument piping has been replaced this outage. Visual inspections were performed of class I piping downstream of these valves. The inspector questioned the basis for licensee confidence in instrument line isolation valve positions during the test. The licensee pointed out that hydrostatic testing of these lines was not required during this outage. In addition excess flow check valve testing was conducted immediately after completion of the hydrostatic test with the system still pressurized. Successful completion of the check valve testing requires proper alignment of the manual isolation valves, and provides assurance that the piping was pressurized during the visual inspections. The licensee however, agreed that the intent of the test had been to pressurize and inspect this piping and that the current procedure does not adequately assure the correct valve alignment. Licensee management stated that the procedure would be revised to address this weakness.

6.0 Integrated Leak Rate Testing

On December 21, 1987, the licensee began performance of the primary containment integrated leak rate test (ILRT). The containment was pressurized with air to the full test pressure of 45 pounds per square inch and maintained at this pressure for 24 hours. The 24 hour test period started at 10:15 p.m. on December 21, 1987. A regional specialist inspector was onsite during the ILRT to review the adequacy of the test procedure and to observe the conduct of the test. The preliminary licensee test results indicated a successful test, with measured leakage slightly greater than 20 percent of the allowable leakage. A primary contributor to the observed leakage was identified as a drywell pressure transmitter piping cap which had not been fully tightened. Upon completion of the specialist inspector's review of the ILRT results, inspection report 50-293/87-58 will be issued documenting the inspectors findings.

While preparing for the primary containment integrated leak rate test (ILRT) the licensee observed that several torus temperature and moisture elements were not functioning properly. Troubleshooting identified circuit faults at a torus electrical penetration assembly. The licensee removed the penetration assembly protective cover inside the torus and found that it was filled with water. The penetration is installed vertically through the top of the torus. On both the inboard and outboard sides of the penetration a metal frame is attached on which 28 terminal boards are mounted. Cables passing through the penetration, and supplying instrumentation in the torus also landed on these terminal boards. A protective cover is bolted over the frame and terminal boards on both sides of the penetration. Design drawings specify that cover joints are to be sealed with silicone tape. The licensee stated that the protective cover had not been properly sealed, allowing water intrusion and buildup. The water caused significant corrosion of the cable connectors, terminal boards and metal framework. This corrosion and water buildup resulted in the observed electrical circuit faults. Licensee inspection of the other torus electrical penetration identified similar conditions. Temporary repairs of the temperature and moisture elements were made to allow ILRT performance. Cables for communications, lighting, and torus to drywell vacuum breaker indication also run through the penetration. The penetration is not considered by the licensee to require environmental qualification but is designated as a "Q" component. The licensee is evaluating the root cause of the water intrusion and is developing a temporary procedure to control repair and testing of the penetration. The inspectors will continue to monitor licensee followup and corrective actions.

The licensee informed the inspector that penetration repairs would not be completed until after ILRT performance. The inspector questioned the effect of the planned repairs on the penetration leak tightness, and the ability to perform adequate leakage test after the planned rework. The licensee stated that the work would not affect penetration leakage but that adequate testing could be performed after work completion. Based on available drawings however, the licensee could not demonstrate adequate testability. In response to NRC concern the licensee obtained the needed drawings from the vendor and verified that the penetration was completely testable. The inspector had no further questions.

During the ILRT, the licensee identified a water leak in the high pressure coolant injection (HPCI) turbine room. It was determined that the increasing pressure in the torus air space caused the suppression pool water to back up through the HPCI turbine exhaust line and through the drain piping, overflowing the HPCI gland seal condenser onto the HPCI room floor. The turbine exhaust line discharges to the torus through a check valve and a locked open stop-check valve. To prevent any condensation from collecting in the turbine exhaust line downstream of the check valve, a drain piping drains any condensation to the HPCI gland seal condenser through a drain pot. Two solenoid operated drain valves on the drain pot close automatically on a HPCI (Group IV) isolation signal. This is to provide the isolation from the torus to the gland seal condenser. The licensee's investigation determined that leads had been lifted in the HPCI isolation interlock logic circuit since October 30, 1987 in support of the HPCI steam testing utilizing temporary oil-fired auxiliary boilers. With the HPCI isolation signal bypassed, the drain valves remained open as the drain pot was filled with the suppression pool water. The licensee subsequently relanded the leads in the HPCI isolation interlock logic circuit and the drain valves closed.

After reviewing the ILRT procedure, HPCI test procedure and interviewing licensee personnel, the inspector concluded that licensee review of the active maintenance requests prior to the ILRT was not thorough in that the lifted leads controlled by the MR 87-663 were not identified. The MR tags were attached on the HPCI isolation logic circuit inside a logic panel and thus the tags were not identified during a system walkdown prior to the ILRT. The drain valve positions were verified by the light indications on the control room panel 903 as prescribed in the ILRT procedure.

The inspector also determined that the maintenance request above may not be an adequate method of identifying and tracking jumpers and lifted leads, especially for a long term application and for components which could affect other ongoing maintenance or surveillance. Station procedures do not require temporary modification controls for jumpers and lifted leads which are controlled by active maintenance requests. The inspector discussed these findings at the exit interview with licensee management. The licensee informed the inspector that a lifted leads and jumper log will be kept in the control room to aid the operators in controlling lifted leads and jumpers.

7.0 Licensee Nuclear Organization Management Realignment

On December 14, and on December 31, 1987, the Boston Edison Co. announced, as part of a planned realignment occurring over the next several weeks, the appointment of the following managers to key management positions in the licensee nuclear organization at Pilgrim Station.

-- Mr. Kenneth L. Highfill was named to assume the new position of Station Director. In this capacity, Mr. Highfill will oversee day to day operation of the Pilgrim Station including plant operations, planning and outage, nuclear training, plant support functions, and administrative services. Mr. Highfill will report directly to Mr. Ralph G. Bird, Senior Vice President-Nuclear.

-- Mr. Robert J. Barrett was named the new Plant Manager. Mr. Barrett will report to Mr. Highfill, the Station Director.

-- Mr. Roy Anderson, currently Deputy Outage Manager, was named to assume the new position of Planning and Outage Manager. Mr. Anderson will report to Mr. Highfill, the Station Director.

-- Mr. Ed Kraft was named to assume the new position of Plant Support Manager. In this capacity, Mr. Kraft will oversee radiological, security, industrial safety and fire protection, and other station support functions. Mr. Kraft will report to Mr. Highfill, the Station Director.

-- Mr. Donald Gillespie, currently Director of Planning and Restart, was appointed to the position of Quality Assurance Department Manager. Mr. Gillespie will assume the position after completing his Senior Reactor Operator training. The Quality Assurance Department Manager reports to Mr. J. E. Howard, Vice President-Engineering.

-- Mr. Frank Famulari, currently Operations Quality Control Group Leader, was named to assume the newly created position of Deputy Quality Assurance Department Manager. Mr. Famulari will report to Mr. Gillespie, and be acting Department Manager until Mr. Gillespie assumes the position after completing the Senior Reactor Operator training.

-- Mr. John F. Alexander was named to assume the position of Operations Section Manager. Mr. Alexander will report to Mr. Barrett, the Plant Manager.

-- Mr. Donald J. Long was named Security Section Manager. Mr. Long will report to Mr. Kraft, the Plant Support Manager.

8.0 Management Meetings

At periodic intervals during the course of the inspection period, meetings were held with senior facility management to discuss the inspection scope and preliminary findings of the resident inspectors. On January 26, 1988, the inspectors conducted a final inspection exit interview to formally present inspection findings.

Attachment I to Inspection Report 50-293/87-57

Persons Contacted

* R. Bird, Senior Vice President - Nuclear
* K. Highfill, Station Director
K. Roberts, Plant Manager
R. Barrett, Deputy Plant Manager
R. Anderson, Planning and Outage Manager
E. Kraft, Plant Support Manager
F. Famulari, Deputy Quality Assurance Manager
D. Swanson, Nuclear Engineering Department Manager
J. Alexander, Operations Manager
N. Brosee, Maintenance Manager
J. Jens, Radiological Protection Manager
J. Seery, Technical Manager
R. Grazio, Field Engineering Manager
P. Mastrangelo, Chief Operating Engineer
R. Sherry, Chief Maintenance Engineer
N. Gannon, Chief Radiological Engineer
D. Long, Security Manager
F. Wozniak, Fire Protection Manager

*Senior licensee representatives present at the exit meeting.

ATTACHMENT II

January 6, 1988

MEMORANDUM FOR: Ken Roberts
Plant Manager

FROM: Clay Warren
Senior Resident Inspector - Pilgrim

SUBJECT: FACILITY TOUR FINDINGS, DECEMBER 8, 1987

The items on the attachment were noted during the facility tour on December 8, 1987. Please contact the Resident Inspector Office when your staff is ready to discuss the evaluation of the items and the status of any actions taken. Please note the items and the facility response will be addressed in a routine inspection report.

Thank you for your time and attention to these matters.

Sincerely,

Clay C. Warren
Senior Resident Inspector

Attachment:
As stated

cc w/attachment:
R. Blough
W. Kane
W. Russell
J. Wiggins

ATTACHMENT

-- Numerous motors appear to have failed grease seals caused by overgreasing without first removing grease drains. This condition causes a buildup of grease and dirt in the cooling airflow path and in extreme cases grease in the motor windings. (SBGT fans and SLC pumps)

-- Nuts and bolts were noted laying inside an electrical cabinet in the RCIC room.

-- Multiple cases of open junction boxes, terminal boxes and conduit pulled away from terminal boxes were noted.

-- Motor heaters for the "B" RHR pump appear to have overheated causing the insulation on the heaters to melt.

-- HPCI room cooler drip pan is full of paint scrappings which could lead to drain clogging.

-- Standby Liquid Control system relief valves have boric acid crystal buildup which could alter setpoints.

-- Painting effort should be more closely controlled to prevent painting inappropriate surfaces, i.e., linkages, valve packing glands, trip throttle valves, limit switches, etc.

-- Numerous instances of scaffolding materials, i.e., nails and wood chips, laying on floors. This material could migrate to drain systems and cause pump or valve damage. Scaffolding was also noted attached to permanent equipment such as piping and conduit.

-- Valve 1001-36A motor operator conduit had melted plastic cover.