U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-293/85-17

Docket No. 50-293

License No. DPR-35 Category C

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

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: June 13, 1985 - July 15, 1985

Inspectors: for L. Tripp 8/1/85
J. Johnson, Senior Resident Inspector Date

for L. Tripp 8/1/85
M. McBride, Resident Inspector Date

Approved By: L. Tripp 8/1/85
L. Tripp, Chief, Reactor Projects Section 3A Date

Inspection Summary: Inspection on June 13 - July 15, 1985 (Report No. 50-293/85-17)

Areas Inspected: Routine unannounced safety inspection of plant operations including: Followup of previous inspection findings and NRC Bulletins, operational reports, ESF walkdowns, surveillance and maintenance activities, and health physics activities. The inspection involved 98 inspection-hours by two resident inspectors.

Results: Two violations were identified (Failure to establish and implement surveillance procedures, detail 7; and Failure to follow a procedure for radiation work permits, detail 8). One deviation from a licensee commitment concerning the Inservice Test (IST) Program was also identified (detail 6). No other significant safety issues were identified.

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DETAILS

1. Persons Contacted

Within the report period, interviews and discussions were conducted with members of the licensee and contractor staff and management to obtain information pertinent to the inspection.

2. Plant Status

The plant operated at full power during most of the inspection period. A reactor scram from low power occurred on June 14, 1985 from a high water level indication during low power maneuvers. Power had to be reduced below fifty percent for several days at the end of June due to heavy fouling of cooling water intake screens during stormy weather.

3. Followup on NRC Bulletins and Previous Inspection Findings

(Closed) IE Bulletin (80-25). Operating Problems with Target Rock Safety-Relief Valves (SRV) at BWRs. This Bulletin was issued to inform all BWR licensees about problems experienced with the two-stage SRVs at Pilgrim in 1980 and to request action regarding 1) inspection of solenoid actuators and testing following an aging period of about 3 months at operating temperatures, 2) revision of procedures to require disassembly, inspection, and testing with steam in the event of failure without clear identification of the cause, and 3) review of the pneumatic supply for the potential for overpressurization, and 4) placement of relief valves and annunciators.

The licensee provided their initial response to this Bulletin in a letter dated March 19, 1981. The following LERs have also been issued by the licensee describing these events and corrective actions: LERs 80-30, 80-47, 80-69, 80-79, 80-80 and 81-62. NRC Reports documenting event followup include the following: Nos. 80-25, 80-26, 80-29, 80-30, 81-24, 81-35, 82-10 and 82-16.

In addition, a recent NRC Inspection, No. 84-39, documents verification of licensee's actions to prevent overpressurization of the nitrogen supply system as described in the licensee's letter dated December 4, 1984.

The inspector also verified that the current station maintenance procedure (3.M.4-6 Rev. 12) continues to contain the requirements for disassembly, inspection, and testing with steam in the event of failure without clear identification of the cause. The Pilgrim station technical specifications were also revised (Sections 3.6.D and 4.6.D) to provide further assurances of proper operation of the SRVs.

The inspector determined that the concerns raised by this Bulletin have been adequately addressed. This Bulletin is closed.

(Closed) Follow Item (84-26-02). Followup on licensee action to resolve discrepancies in vessel water level needed to maintain 2/3 core coverage. The licensee submitted a proposed technical specification change, dated June 18, 1985, to correct the 2/3 core water level setpoint listed in T.S. Table 3.2.B. The setpoint prevents the diversion of residual heat removal (RHR) system water to containment sprays when the water is needed for low pressure coolant injection (LPCI) to maintain 2/3 core coverage. The setpoint was changed from 302 inches above vessel zero to 307 inches above vessel zero. The licensee submittal indicates that the change is an administrative matter and does not impact on plant safety. The inspector had no further questions at this time. This item is closed.

(Closed) Violation (84-26-04). Failure to properly review and approve QA-related procedures. The inspector verified that four procedures identified in the violation had been subsequently reviewed and approved by the QA Manager. The following additional corrective actions were verified:

- The licensee completed an initial review of QA program-related procedures by January 29, 1985. The review identified additional procedures which needed QA approval. The approval of the additional procedures was tracked by the QA Department.
- The licensee index for QA program-related procedures was updated by March 1, 1985 to incorporate the results of the January review.
- A requirement to annually review issued procedures against the index was incorporated into QA procedure 5.03.
- Station procedure no. 1.3.4, "Procedures", was revised to require that QA program-related procedures listed in the index be sent to the QA Manager for review and approval.

The inspector had no further questions concerning QA approval of procedures at this time. This item is closed.

(Closed) Unresolved Item (85-16-02). Review implementation of the Inservice Testing Program. This item is closed for administrative purposes and will be tracked under the item of deviation identified in Detail 7 of this report.

4. Operational Safety Verification

a. Scope and Acceptance Criteria

The inspector observed control room operations, reviewed selected logs and records, and held discussions with control room operators. The inspector reviewed the operability of safety-related and radiation monitoring systems. Tours of the reactor building, turbine building, intake structure, station yard, switchgear rooms, battery rooms, and control room were conducted.

Observations included a review of equipment condition, security, housekeeping, radiological controls, and equipment control (tagging).

b. Findings

(1) On June 18, 1985, the indicator for the "B" Yarway reactor water level indicator in the control room decreased to about +18 inches. This instrument shares common indicating and reference lines with safety related level switches for the reactor protection, emergency core cooling, and containment isolation systems. Water level is usually maintained between +24 and +28 inches. Other level instrumentation in the control room did not confirm a decrease in water level. The "B" Yarway indicator returned to normal within a few minutes. The licensee subsequently checked the instrument calibration but could not identify any problems. A signal conditioning board was replaced in the unit.

On July 14, 1985, the "B" Yarway indicator drifted upwards to +38 inches from an initial value of +28 inches. As before, other control room level instruments did not confirm a decrease in reactor water level and the Yarway returned to normal level within a few minutes.

The licensee stated at the exit interview that personnel located at the local instrument racks in the reactor building did not observe the indicators safety related level switches drifting during the transient on July 14. Personnel did not observe the switches during the first transient on June 18. The inspector had no further questions at this time. The stability of the Yarway reactor water level instruments will be reviewed during future routine inspections of control room activities.

(2) On June 24, 1985, the "A" loop of the core spray system was made inoperable in order to upgrade system components for environmental qualification. The inspector verified that the appropriate procedure (No. 8.5.1.4) for one inoperable core spray subsystem was initiated and that other emergency cooling system surveillance tests were conducted as required by the technical specifications. The "A" core spray system was placed back in service on June 29, 1985. No inadequacies were identified.

(3) On June 24, 1985, the secondary containment system isolated on a false refuel floor exhaust duct radiation signal. The licensee promptly reset the isolation. An investigation indicated that a licensed operator may have inadvertently caused the false radiation signal by jarring the radiation instrument drawer in the control room while conducting a functional test of the radiation monitors. No inadequacies were identified.

(4) On June 25, 1985, the inspector noted that the "A" logic for group I primary containment isolation was tripped. The tripped logic was indicated on a back panel in the control room and the control room operators were not aware of the half isolation at that time. The signal was promptly reset.

The licensee determined that the isolation was caused by an electrical transient in a 480 V a.c. bus (B-17). This bus feeds two 120 V a.c. buses, Y-3 and Y-31, which power a portion of the primary containment isolation system logic. Activation of additional isolation logic which was powered by Y-3 and Y-31 was noted at that time.

The transient in B-17 occurred when a breaker sparked during post work testing on a core spray valve MO-1400-3A. The licensee evaluated the condition of B-17, Y-3, and Y-31 after the transient and found no damage. All fuses in Y-3 and Y-31 were checked and found acceptable. The inspector had no further questions at this time.

(5) On June 29, 1985 at 6:31 p.m., both reactor recirculation loop flows unexpectedly increased while the reactor was at full power. Reactor pressure increased to 1045 psig and power rose to 2034 Mwt (101.8% of the steady state power limit). The recirculation speed demands for both loops were promptly lowered and power and pressure returned to normal.

The recirculation loop flows should be independent. While a master controller for both loops is available, this controller has not been used during the current operating cycle because of slight flow stability problems in the "A" loop. Instead, each recirculation loop has been controlled with an independent controller. The licensee evaluated the flow controllers after the transient, but could not identify any problems.

On July 1, 1985 at approximately 11:00 a.m., flow in both recirculation loops, reactor pressure and reactor power unexpectedly increased for a second time. Pressure increased from 1032 to 1036 psig and power rose from 1990 to 2012 Mwt. As before, recirculation speed demands were promptly decreased and pressure and power returned to normal.

No causes for the transients were identified by licensee personnel on duty during the transients. In response to questions from the inspector, the licensee is conducting a more formal evaluation of the incidents. No further transients involving both recirculation loops occurred during the inspection period. The inspector had no further questions at this time. No inadequacies were identified at this time. This is being carried as an Open Item (85-17-05).

(6) On June 30, 1985 at 6:30 a.m., one of two rupture diaphrams in the steam exhaust line for the high pressure coolant injection (HPCI) system developed a leak during a routine monthly pump surveillance test. The licensee declared the HPCI system inoperable at 11:55 a.m. on June 30, 1985, following a check of diaphram leak detection instrumentation and a second HPCI test. The licensee notified the NRC of the HPCI problem via the ENS telephone line at 12:45 p.m.

The HPCI system was declared operable at 5:10 p.m. on June 30, 1985, after the diaphram was replaced and the system successfully run. The licensee stated that the diaphram damage was slight and probably caused by fatigue from repetitive overspeed transients in early June (NRC Special Inspection 50-293/85-16). The diaphram was previously replaced following a HPCI trip and water hammer event in March, 1985 (NRC Inspection 50-293/85-08). The inspector had no further questions at this time. Future HPCI surveillance tests will be reviewed during routine inspections of the licensed program.

(7) On July 12, 1985 at 3:00 p.m., a spurious secondary isolation occurred during a routine isolation logic surveillance test. The isolation signal was generated when a technician opened the wrong relay contacts at the start of a test of the "B" containment isolation logic train, Procedure No. 8.M.2-1.5.8.2. The Chief Maintenance Engineer indicated that the technician did not follow the test procedure and assumed that the same relay contacts were opened in the "B" logic train test as in the "A" logic train test.

The logic test was immediately terminated and the containment isolation reset. The NRC was notified of the isolation via the ENS telephone line at 4:58 p.m.. A maintenance supervisor verified that the isolation logic was left in an acceptable state after the terminated test on July 12, 1985.

The technician was counselled on the importance of closely following procedures. In addition, all instrument technicians (Nuclear Control Technicians) were informed of the incident and the root cause. An inspection of secondary containment dampers was also initiated.

The Chief Maintenance Engineer indicated that the technician involved in the incident was one of the most experienced technicians in the station and had done the surveillance test many times. Surveillance test no. 8.M.2-1.5.8.2 was subsequently completed on July 13, 1985. The licnesee plans to submit an LER on the event. The inspector had no further questions at this time.

5. ESF Walkdown

a. Scope

On July 8 and 9, 1985, the inspector walked down portions of the safety related 125 and 250 V d.c. systems. The electrical lineups were compared to station drawings. Equipment maintenance and surveillance testing were also reviewed.

b. Findings

(1) An apparent violation of the technical specification involving surveillance testing of station batteries is discussed in Section 7 of this report.

(2) The backup battery charger for the 125 V d.c. system has been out of service since July, 1984. This charger is not required to be operable by the technical specifications. The licensee believes that damaged printed circuit boards in the charger caused it to fail tests in 1984. However, the charger could not be repaired subsequently using circuit boards that the licensee had in stock. The licensee has received additional boards from the charger manufacturer and plans to install these in the unit in the near future.

At the exit interview, the licensee stated that a safety evaluation was not required to keep the backup charger out of service. The licensee also indicated that differences between the recently received circuit boards and the older boards in the licensee's stock would be evaluated. The inspector had no further questions concerning the charger at this time.

(3) The inspector noted that breaker no. 21 in a 125 V d.c. distribution panel, D6, was tagged in the open position. This breaker feeds an emergency station lighting panel, 25L. A nuclear Watch Engineer's tag, placed on the breaker in October, 1984, indicated that a QC nonconformance report (NCR) had been issued against a component in the lighting bus. Licensee records indicated that an NCR had been issued and cleared during the 1984 plant outage. Quality Control personnel stated that no NCRs were currently open on the 25L panel.

The licensee promptly removed the tag and energized the breaker.

Personnel from the licensee engineering staff indicated that the emergency lighting on the 25L panel was not used to fulfill the emergency lighting requirements of 10 CFR 50 Appendix R. The licensee stated at the exit interview that QC personnel had not informed the control room when the NCR on the 25L bus had been cleared. The licensee stated that QC had been reminded to tag control switches as well as local components in the future.

The inspector reviewed the startup checklist for the startup from the 1984 outage and verified that the licensee had noted the tag during a system walkdown. The inspector had no further questions at this time.

6. Followup on Events and Nonroutine Reports

a. Events

On June 15, 1985 at 4:35 a.m., the reactor scrammed during maneuvers at about 700 psig reactor pressure. Reactor power had been previously reduced in preparation for repairs to the turbine control oil system.

The scram was caused by an automatic closure of the main steam isolation valves on a high reactor water level signal at greater than 600 psig reactor pressure due to operator error. The licensee conducted a post trip review and checked the calibration of safety related reactor water level instrumentation. No inadequacies were identified during the review or calibration checks. The reactor was maintained in hot shutdown until later that day when a reactor startup was initiated.

The inspector observed control room activities while the reactor was shutdown. The high pressure coolant injection system (HPCI) was used in the test mode (recirculate to the condensate storage tank) to draw off steam and cool the reactor. Water was supplied to the reactor via the feedwater system. The HPCI system started smoothly and functioned normally. The inspector also reviewed the level instrument calibration results, procedure no. 8.M.1-19. No inadequacies were identified.

b. Review of Licensee Event Reports (LERs)

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

No.	Subject
85-12	HPCI System Inoperable (LERs 85-12-00 and 85-12-01)
85-13	HPCI Isolation
85-14	Reactor Scram

The two HPCI inoperable events described in LER 85-12 were reviewed during NRC inspections 50-293/85-11 and 85-16. The inspector noted that the cause code for the second HPCI inoperable event was not given in

either the original or revised LER 85-12. The licensee agreed to review the second event and submit a revised LER if an additional cause code is needed.

The HPCI isolation described in LER 85-13 was reviewed during NRC inspection 50-293/85-11. The reactor scram in LER 85-14 is discussed in Section 6 of this report.

The inspector had no further questions concerning LERs at this time.

7. Surveillance Testing

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

A list of items reviewed is included at the end of this report in the attachment to this report.

b. Findings

(1) During a review of station battery surveillance test check lists for May, June and July, 1985, the inspector noted one instance where a weekly test of specific gravity, voltage, and temperature had not been conducted for a pilot cell in the station 250 V battery. The battery was properly tested on May 26 and June 9, 1985, but was not tested during the routine weekly surveillance on June 2, 1985.

The surveillance on the 250 V battery was missed on June 2, 1985 because the procedural check list, no. 8.C.14, did not require that the battery be tested. Instead, the check list required a duplicate test on another battery, the 24 V (A) battery. The licensee stated that the 250 V battery test was inadvertently left off of the checklist when the list was revised in April, 1985. Most of the weekly tests since then had the duplicate 24 V battery test crossed out on the check list and in its place had a 250 V battery test handwritten on the form. However, the procedure was not formally changed.

The following instances of failure to follow surveillance Procedure 8.C.14 were also noted:

-- Battery temperatures were logged during June 9 and 23, 1985 tests which slightly exceeded the procedural acceptance criteria of 77 ±15 degrees Fahrenheit. An Operations Supervisor and a Watch Engineer had approved the completed tests on both dates. The increase in battery temperatures on these dates did not appear to indicate battery problems, but rather elevated ambient room temperatures.

On June 9, 1985, operations personnel may not have realized that the acceptance criteria had been exceeded because a centigrade thermometer was used instead of a Fahrenheit thermometer. Procedure 8.C.14 did not contain acceptance criteria expressed in centigrade units. At the exit interview, the licensee stated that Procedure 8.C.14 would be modified to include both Fahrenheit and centigrade acceptance criteria.

-- On June 9, 1985, voltage levels for the D17-125 V and D10-250V buses were not entered on the check list as required. On July 9, 1985, 230 V was entered as the D17-125 V battery voltage. Battery voltages entries required by the technical specifications were indicated elsewhere on the check lists for both dates (expressed as voltages across the appropriate battery chargers).

-- On July 6 and 13, 1985, cells with the lowest specific gravities in the 125 (b) V and 250 V batteries were not chosen pilot cells for testing as required by Procedure 8.C.14.

Technical Specifications 4.9.A.2 requires that a pilot cell in the 250 V battery be tested for specific gravity, voltage, and temperature at a frequency of once a week. Failure to establish a procedure to implement this test is a violation of Technical Specification 6.8. Failure to follow the instructions in Procedure 8.C.14 on the dates indicated above is also a violation of Technical Specification 6.8 (85-17-01).

(2) On June 24, 1985, the licensee indicated that one of the pump injection valves, MO-2301-8, for the high pressure coolant injection (HPCI) system had not been submitted to the NRC. The inspector had previously questioned the adequacy of the testing, after noting that the IST program required the stroke time of the valve to be measured in both the open and close directions but the licensee's test procedure only timed the valve in the open direction.

The inspector requested that the licensee review the IST program to ensure that all tests were implemented. During this review, the licensee noted failures to accomplish the following testing:

System	Valve	Test	Required Frequency
HPCI	2301-8	Stroke time to close position	Quarterly
	VRV-9066	Relief valve setpoint verification	5 years
	VRV-9066	Stroke to open position	Quarterly

Diesel Oil Trans.	1 in. No. 223 (2 valves)	Stroke to close position	Quarterly
	Foot-VAL A & B	Stroke to close position	Quarterly

The licensee indicated that the stroke time to close position test for the 2301-8 valve would be added to the routine surveillance procedures. The HPCI relief valve tests will be reviewed by the licensee's engineering department to determine whether the IST tests are feasible. The licensee will also review the diesel oil transfer system to determine if the IST tests are feasible.

At the exit interview, the inspector noted the importance of the HPCI relief valve in preventing water hammers in the HPCI steam exhaust line. Two water hammer events have occurred this year (LERs 85-08 and 85-12). The licensee indicated that a similar vacuum relief valve in the reactor core isolation cooling (RCIC) system has also not been tested. The RCIC valve is not listed in the IST program as this system is not safety related.

The licensee indicated that both the HPCI and RCIC vacuum relief valves were welded into place and could not be easily removed for testing. The licensee may install flange connections for the valves in the future. Setpoint testing was considered during the last outage, but was not done. Neither valve has been tested since the time of original installation.

The licensee stated that the original IST program submitted to the NRC, dated April 13, 1979, required the HPCI relief valve set point to be tested every five years. The second 10-year IST program submitted to the NRC, dated July 11, 1983, and a revision to the program, dated February 14, 1984, also require that the valve be tested. Both the original and revised second 10-year IST programs require that the other valves listed in above table be tested as indicated. Failure to perform these tests is a deviation from an NRC commitment (85-17-02).

8. Maintenance and Modification Activities

a. Scope

The inspector reviewed the licensee's actions associated with maintenance and modification activities in order to verify that they were conducted in accordance with station procedures and the facility Technical Specifications. The inspector verified for selected items that the activity was properly authorized and that appropriate radiological controls, equipment tagging, and fire protection were being implemented.

A list of the items is included at the end of this report in the attachment to this report.

b. Findings

On July 10, 1985, the inspector discussed recent information concerning the environmental qualification of Limitorque motor operators with the Chief Maintenance Engineer (CME). Tests have indicated that magnesium rotors in some operators may quickly corrode if exposed to a steam environment. The CME stated that General Electric had been asked to determine if any of the Limitorque operators installed at Pilgrim have the magnesium rotors. The CME indicated that the test data for the magnesium rotors may be misleading and would be evaluated further. The results of the General Electric and licensee evaluations will be reviewed during a future inspection (85-17-03).

9. Health Physics Activities

a. On July 3, 1985, the inspector noted that the frequency of periodic radiation surveillance was not clearly indicated on licensee radiation work permits (RWP) for high radiation areas, as required by station procedures and the technical specifications. For example, RWPs issued for work on the -13 ft level of the radwaste building and for the radwaste trucklock indicated that health physics surveillance was to be periodic, but did not indicate a specific surveillance frequency. Radiation levels in areas controlled by these RWPs were up to 450 mR/hr.

The health physics supervisor who authorized the RWPs indicated that he routinely specified "periodic" surveillance coverage for high radiation areas that did not warrant constant coverage. A frequency of "daily when worked" was specified in a "Survey Frequency" box on the RWPs. The supervisor indicated that this frequency meant that health physics technicians had to enter high radiation areas at a minimum frequency of once per shift to provide surveillance coverage and had to conduct a comprehensive radiation survey once per day (24 hours). However, two of four health physics technicians interviewed during the inspection understood the "daily when worked" phrase to mean a minimum radiation surveillance frequency of once per 24 hours.

The Chief Radiological Engineer (CRE) indicated that the "Survey Frequency" block was only to be used to indicate the frequency of comprehensive surveys, not the frequency of high radiation surveillances. Procedure No. 6.1-022, "Radiation Work Permit", requires that both the survey frequency and the surveillance frequency be entered on RWPs. In addition, the CRE issued written instructions to his staff in February 1985, which required both the survey and surveillance frequencies to be specified on RWPs.

In response to this finding, the licensee revised all RWPs for high radiation areas changing periodic surveillance coverage to a once per frequency shift. The licensee also discussed the need to indicate a surveillance frequency on RWPs with all health physics supervisors who authorize the RWPs and with the station health physics technicians.

Concerns about the adequacy of RWP control of high radiation area surveillance were discussed with the licensee in connection with the first "chips" incident in 1984 and documented in NRC report 50-293/85-03. Additional concerns were raised during a subsequent radiation specialist inspection and documented in Report 50-293/84-25. The need to ensure that high radiation surveillance coverage is clearly specified on RWPs was discussed in meetings with licensee management in March 1985 and documented in report 50-293/85-06.

Failure to specify a surveillance frequency on RWPs controlling individuals who do not have radiation monitoring instruments or alarming dosimeters and who enter high radiation areas is a violation of Technical Specification No. 6.13 and licensee Procedure No. 6.1-022 (85-17-04).

b. The following information is included in this report to assist NRC management in following radiation exposure at the station. The monthly personnel radiation exposure for June, 1985 was 66.7 person-rems. The total yearly exposure through June 30, 1985 was 422.1 person-rems.

10. Management Meetings

During the inspection, licensee management was periodically notified of the preliminary findings by the resident inspectors. A summary was also provided at the conclusion of the inspection and prior to report issuance. No written material was provided to the licensee during this inspection, other than documents available in the public document room.

ATTACHMENT TO INSPECTION REPORT 85-17

The following surveillance and maintenance items were reviewed during the inspection period.

a. Portions of the following tests were reviewed:

- -- Reactor pressure permissive instrument calibration on June 16, 1985.
- -- Scheduling of ECCS surveillance tests between June 23 and July 1, 1985 while the "A" loop of the core spray system was inoperable.
- -- Secondary containment damper inspections on June 23 and July 1, 1985.
- -- Rod block monitor functional test on June 25, 1985.
- -- Local power range monitor calibrations on July 2, 1985.
- -- Weekly tests of station batteries during May, June, and July 1985.
- -- Quarterly battery test on July 2, 1985.
- -- ECCS and diesel generator surveillance tests on July 7 and 8, 1985 prior to making the "B" loop of core spray inoperable.

b. Portions of the following maintenance and modification activities were reviewed:

- -- MR 85-24-96, AON-90 has a cracked drive gear.
- -- Temporary Modification 85-34, repair drive louver on AON-90.
- -- MR 85-430, HPCI exhaust diaphram high pressure annunciator.
- -- MR 85-431, replace HPCI exhaust diaphram.
- -- MR-46-426, low specific gravity L185.
- -- Deficiency Tags 2290 and 2333 on 250 V d.c. distribution panel
- -- Maintenance on the backup 125 V station battery charger.