U.S. NUCLEAR REGULATORY COMMISSION DCS Nos. 50293-850331
REGION I 50293-850404

Report No. 50-293/85-08

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: April 2, 1985 - May 6, 1985

Inspectors: for J. E. Tripp 5/28/85
J. Johnson, Senior Resident Inspector Date

for J. E. Tripp 5/28/85
M. McBride, Resident Inspector Date

Approved By: J. E. Tripp 5/29/85
L. Tripp, Chief, Reactor Projects Date
Section No. 3A, Projects Branch No. 3

Inspection Summary: Inspection on April 2, 1985 - May 6, 1985 (Report No. 50-293/85-08)

Areas Inspected: Routine unannounced safety inspection of plant operations including: Followup on previous inspection findings, operational safety verification, followup of events and non-routine reports, surveillance and maintenance activities, chemistry and health physics activities, and a review of licensee actions regarding the use of liquid nitrogen in containment systems. The inspection involved 134.5 inspection-hours by two resident inspectors.

Results: No violations were identified.

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DETAILS

1. Persons Contacted

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

2. Plant Status

   The plant operated at full power during most of the inspection period. A reactor scram occurred on April 4, 1985 following a trip of the main turbine. Power operation was resumed on April 5, 1985.

3. Followup on Previous Inspection Findings

   a. (Closed) Violation (293/83-09-01). Safety related material was accepted for use during receipt inspection without meeting the documentation and identification requirements of Purchase Order 62102. NRC Report No. 83-19 documents initial review of this item and the implementation of corrective actions described in the licensee's response dated July 6, 1983. The inspector also verified that the Quality Assurance Manual, Section 15.2.5, requires the Nuclear Engineering Department and Quality Assurance Department to review and approve "accept" or "repair" dispositions to Nonconformance Reports (including those written for receipt inspection). This item is closed.

   b. (Closed) Follow Item (293/83-24-02). Check that acceptance criteria are included in the diesel generator fuel oil surveillance procedure. The inspector reviewed procedure no. 7.1.3.6, "Diesel Generators' Fuel Oil Sampling and Quality Analysis", Revision 4, and noted that ASTM acceptance criteria are included in the procedure. The inspector had no further questions. This item is closed.

   c. (Closed) Unresolved Item (293/84-24-01). Clarify inspection requirements on work plan and inspection record (WP & IR) documents. The WP & IR procedure, no. WP/P-1, was revised following the 84-24 inspection to include additional instructions for documenting WP&IR inspection requirements. The inspector reviewed the revised procedure and a WP & IR for installing hangers for electrical conduit, No. 84-060.05-E-152. No unacceptable conditions were identified. Additional WP & IR's will be reviewed during future routine inspections of the licensed program. This item is closed.

   d. (Closed) Violation (293/84-36-01). Failure to maintain the "B" source range monitor (SRM) operable during fuel movement in that quadrant. The licensee's response to the violation, dated January 9, 1985, stated that a "Fuel Load Checklist" would be added to operator shift turnover sheets to help ensure that information such as bypassed SRM's would be transmitted between shifts. The inspector verified that the checklist had

been added to the turnover sheets and noted that operators appeared to be adequately aware of the state of nuclear instrumentation during the remaining refueling activities in 1984. The inspector had no further questions. This item is closed.

e. (Open) Unresolved Item (293/85-06-03). Core spray recirculation test valve, MOV 1400-4A, operator mounting cap screws found loose. Following this latest finding (LERs 83-010 and 83-035 report earlier problems), the station again requested the Nuclear Engineering Department (NED) to evaluate the cause of the loose screws. The inspector discussed the recommended corrective actions with the Station Chief Maintenance Engineer. NED has recommended that a minor modification be made to the mounting screws (drill holes in the screw heads) and lockwire the four screws to prevent their coming loose. Previous efforts to use a thread compound and increased torque were not effective. The licensee is also continuing to evaluate system vibration. The licensee's planned actions include an inspection of the valve after each operation. The recommended modification to lockwire the screws is pending availability to take the A core spray out of service for environmental qualification upgrade replacements. This item remains open pending further review of the licensee's corrective actions.

4. <u>Operational Safety Verification</u>

a. <u>Scope and Acceptance Criteria</u>

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, diesel generator rooms, battery rooms, and control room were conducted.

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

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

b. <u>Findings</u>

(1) On April 5, 1985 at 6:09 am the licensee brought the reactor critical following an unplanned trip the day before. The inspector observed the licensee's actions in the control room during a routine reactor vessel and reactor coolant system heatup to normal operating temperatures and pressures. The inspector noted that the average rate of temperature change was within the T.S. limit of 100 F/hr and that the required temperatures were logged every 15 minutes as required by T.S. 4.6.A.1. However, the inspector questioned the Watch Engineer as to why the temperature logging was stopped at

10:45 am on April 5, 1985 since the heatup of the reactor vessel metal was still in progress and since T.S. 4.6.A.1 requires logging until the difference between two readings in a 45 minute period is less than 5 F.

The licensee stated that the mode switch had been placed in the RUN mode, that the operators had no control over metal temperature rise, and that pressure and temperature control were essentially in automatic. The inspector concluded that since the LCO only applied at vessel temperatures $<$ 450 F and that the required logging was performed up to this point that the licensee's actions were acceptable.

At the exit interview, the licensee acknowledged the inspector's comments regarding confusion that some operators had because of the unclear applicability of T.S. 4.6.A.1 logging requirement above 450 F. At the exit meeting, the licensee stated that a review of the T.S. wording would be performed.

(2) On April 7, 1985 at 11:46 pm, the "F" channel of the average power range monitors (APRM) tripped high, generating a half scram signal. The licensee promptly cleared the half scram and determined that a momentary reactor power spike of three to four percent had caused the "F" APRM to trip. The power spike was believed to be caused by the bistable vortex phenomena in the recirculation system described in NRC inspection report 50-293/85-03. The licensee stated that the combination of a conservative trip setting and a low gain adjustment factor for the "F" APRM had caused it to trip before the other APRM channels. No further APRM trips of this type occurred during the inspection period. No unacceptable conditions were identified.

(3) On April 12, 1985 at 11:55 am, the licensee determined that the refuel floor supply ventilation duct could not be isolated by secondary containment dampers. Specifically, two dampers in series (AO-N-82 and 83) would not properly close during a routine damper inspection. The licensee initiated a reactor shutdown and notified the NRC via the ENS telephone line. The shutdown was secured at 1:46 pm when one of the two dampers was manually closed.

The dampers were subsequently repaired and tested under maintenance requests (MR) No. 85-282 and 85-283. The licensee replaced plastic drive gears on the dampers and repaired a broken damper blade. The dampers were returned to service on April 30, 1985.

Several secondary containment dampers, including AO-N-82 and 83, had some of their drive gears repaired during March, 1985. The licensee indicated that the damper problems were related to the damper design. The secondary containment damper drive mechanisms are scheduled to be replaced during the next refueling outage with

dampers of heavier design. The licensee indicated that the frequency of damper visual inspections will be increased from quarterly to monthly.

The inspector reviewed the secondary containment leakage test results for the current operating cycle and noted that the leakage has markedly increased. In October, 1984, the average differential pressure in the reactor building during a test was -0.36 in. H_2O relative to outside atmosphere. The next test, conducted on March 18, 1985, demonstrated a lower vacuum, -0.27 in. H_2O. The licensee could not explain the increased leakage during the second test and indicated that the next secondary containment leakage test would not be conducted until 1986, during the next refueling outage. The inspector expressed concern over the length of the surveillance interval, in light of the change in secondary containment leakage and the recent damper problems. A subsequent licensee safety evaluation, No. 1812, for a temporary modification of damper AO-N-82 indicated that a secondary containment leakage test may be conducted if future damper problems are found.

The inspector discussed unclear aspects of the secondary containment definition and LCO contained in the plant's technical specifications with the licensee. The licensee indicated that the requirements in the standard BWR technical specifications were generally followed in matters relating to secondary containment. The inspector suggested that the plant's technical specifications be changed to better reflect the licensee's policy for secondary containment. The licensee acknowledged the comment.

The results of future damper inspections and secondary containment leakage tests will be reviewed during routine NRC inspections of the licensed program. The inspector had no further questions at this time.

(4) On April 17, 1985 at 8:30 am, a half scram signal was generated when an I&C technician accidentally shorted two test leads while performing a calibration of local range power monitors (LPRM). The test leads were shorted when the technician inserted the leads into the current measuring test sockets rather than the voltage measuring test sockets on a digital multimeter during the test.

The shorted leads damaged a recirculation flow converter unit for the flow biased circuits on the A, C, and E average power range monitors (APRM). The APRM's then tripped on a false high reactor power with no flow signal.

The technician indicated that he had been distracted by someone talking when he incorrectly connected the test leads to the multimeter.

The inspector reviewed the technician's work experience and concluded that the individual was qualified to perform the LPRM calibration. The incident appeared to be an isolated case of personnel error. The inspector had no further questions.

(5) On April 24, 1985, the inspector noted that the "B" main stack radiation monitor was reading about 30% higher than normal. The licensee reviewed the stack strip chart recording and determined that the monitor output increased in a step change immediately following a source check. The monitor subsequently returned to normal levels.

The licensee checked both main stack detectors and concluded that the increase in the "B" monitor was due to an problem with the source check switch. The licensee stated that the problem was infrequent and that the switch did not need to be repaired. The monitor functioned normally during the remainder of the inspection period. The inspector had no further questions.

5. Followup on Events and Nonroutine Reports

a. Reports

(1) The High Pressure Coolant Injection (HPCI) system had been declared inoperable prior to the beginning of this inspection period (1:50 am on March 31, 1985) because of an overspeed condition caused by a broken electrical connector. Alternative system testing was initiated and continued until the HPCI system was returned to service. This connector was repaired and the HPCI system was tested on April 2, 1985 but was not immediately returned to service because of indication of a ruptured exhaust line diaphragm.

Subsequently, a contractor working in the HPCI quadrant during the evening of April 4, 1985 noted a broken snubber shaft on the HPCI turbine exhaust line and immediately reported this to the control room. This snubber is in the overhead in the room and may have been damaged during previous operation and not noticed.

Maintenance personnel as well as Quality Control personnel performed an inspection of the system. The station also called the Nuclear Engineering Department staff to the plant to evaluate adequacy of the piping support design.

The damaged snubber was rebuilt, retested, and reinstalled. The inspector reviewed the licensee's evaluation of this support, NED 85-338, dated April 6, 1985. This report included an evaluation by the original designer, Teledyne Engineering Services, and recommended restoration of the pipe support, to its original design (with different anchor bolts to ensure proper embedment because of being slightly pulled out of the ceiling). This report concluded that

a water hammer type event had taken place and recommended short term corrective actions to preclude further damage: 1) increase turbine exhaust line nitrogen purge time from 2 to 3 minutes after operation, 2) blowdown the line once per day, and 3) inspect the pipe clamp, snubbers, and baseplate after system operation.

The inspector noted that these actions were being implemented. The control room supervisor was keeping records of exhaust line blowdown done for 3 minutes each night and about midnight. The test procedure (8.5.4.1) was revised to inspect the pipe clamp, snubbers and baseplate after system operation as well as perform a purge of the exhaust line for 3 minutes.

The HPCI system was retested and declared operable at 1:20 am on April 7, 1985. This event is described in LER No. 85-08 dated April 26, 1985. This LER states that additional root cause analysis is in progress and will be discussed in an update to this LER. The licensee is also re-evaluating 1) the test method including advantages and disadvantages of a "quick start" versus a slow start and 2) the capacity of vacuum relief on the exhaust line.

The inspector had no further questions regarding this event at this time. A review of the licensee's root cause analysis and updated LER will be performed in a future inspection (85-08-01).

At the exit meeting the inspector questioned the licensee's rationale for taking the reactor critical on April 5, 1985 (following an unrelated main turbine trip on April 4, 1985) prior to the HPCI system being declared operable. The licensee stated that the basis for this included: 1) the demonstration that redundant safety systems were operable, 2) the requirement of T.S. 3.5.C that the HPCI system be operable prior to a startup from a Cold Condition, and 3) the maintaining of the plant in a Hot Condition. The inspector verified redundant system operability and maintenance of a Hot Condition and had no further questions at this time.

(2) An automatic main turbine trip (and associated reactor scram) occurred from 85 percent power at 11:04 am on April 4, 1985 due to a high vibration signal from a main generator bearing. Control room operators had earlier reduced power from 100 percent following receipt of a vibration alarm at 10:58 am.

The inspector observed plant conditions shortly after the trip and noted that reactor safety systems responded normally and that no ECCS equipment was called upon to operate. Reactor coolant system pressure was 925 psig, all control rods were inserted, and operators were following the shutdown procedure (2.1.5).

The licensee's investigation (with assistance of a G.E. turbine consultant) determined that the vibration sensing probe was not working properly (unevenly worn and insufficient oil flow). The vibration probe was replaced, a generator oil orifice was unplugged, and an unrelated broken selector switch for the "B" source range monitor was repaired.

The reactor was taken critical at 6:09 am on April 5, 1985 and the main turbine was placed back in service at 1:40 pm. The inspector had no further questions regarding this event.

b. Review of Licensee Event Reports (LER's)

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

No.	Subject
85-04	Reactor Vessel Drain Line Leak
85-05	Missed Surveillance Test
85-06	Reactor Scram During Surveillance Test
85-07	Secondary Containment Dampers Inoperative
85-08	HPCI System Inoperable
85-09	Reactor Scram on Turbine High Vibration

The events in LER's 85-04 and 85-05 were reviewed during Inspection No. 85-03. The events in LER's 85-06 and 85-07 were reviewed during Inspection 85-06. Additional failures of the secondary containment dampers described in LER 85-07, AO-N-82 and 83, are described in this report. The events in LER's 85-08 and 85-09 are described in this inspection report. No inadequacies were identified.

6. 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.

b. Findings

(1) The inspector witnessed a test of the High Pressure Coolant Injection (HPCI) system at 1:00 pm on April 2, 1985. This test was being performed to demonstrate system operability following maintenance

(replacement of an electrical connection on the turbine control equipment). The inspector noted that the repaired equipment operated properly and that the test was conducted in accordance with the station test procedure (8.5.4.1). However, the inspector observed an annunciator that alarmed and was acknowledged by the operator during the test, "HPCI turbine exhaust diaphragm hi pressure". The control room operator and supervisor stated that this annunciator had been in alarm during a previous test, that two operators were on station in the torus room and the HPCI room to observe system piping and equipment, and that no reports of exhaust piping problems were observed, and there was no reason to immediately secure the HPCI system (a system shutdown as discussed in the alarm response procedure, 2.3.2.2).

Following discussions with the inspector, the control room supervisor initiated a plant maintenance request and maintenance personnel inspected and replaced the exhaust piping rupture diaphragm. One diaphragm was deteriorated but investigation and checks indicated proper operation of the pressure switches for alarm and HPCI turbine trip. In kind replacement diaphragm rating was stamped as between 180.5 and 199.5 psig, however, the HPCI turbine exhaust pressure trip would occur earlier, at about 150 psig. Exhaust pressure did not reach rupture pressure. LER No. 85-08 postulates that an earlier water hammer event may have caused the problem with the rupture disc. This LER is already being tracked (see IFI No. 85-08-01 in Section 5.a of this report).

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

(2) On April 17, 1985, the inspector observed a calibration of the refueling floor ventilation radiation monitors. These monitors generate a high level trip which isolates secondary containment.

The licensee personnel performing the calibration followed the appropriate procedure, No. 6.5-170. However, the certification documentation for the calibration radiation source they used was not complete. The licensee subsequently verified the calculated dose rates in the calibration source with an "R" chamber detector. The "R" chamber had been calibrated by NBS. The licensee stated that additional documentation for the calibration source would be sought from the source vendor, General Electric. The inspector had no further questions.

(3) On April 24, 1985, the inspector observed a portion of a functional test of the automatic initiation of the auto depressurization system (ADS) on high drywell pressure. The personnel performing the test followed the appropriate procedure, No. 8.M.2-2.1.6. No inadequacies were identified. The inspector had no further questions.

7. 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 reviewed is included at the end of this report in the attachment.

b. Findings

(1) On April 7, 1985, the licensee repaired the AO 220-44 reactor coolant sample line isolation valve. This valve had been repaired and successfully tested in March, 1985, but had subsequently failed closing time tests. The licensee indicated that the valve closing time was decreased on April 7, by modifying the nitrogen bleed off rate from the valve and by lowering the volume of gas in the valve operator.

The nitrogen bleed off was increased by replacing an orifice on the solenoid valve that controls nitrogen flow from AO 220-44. The volume of nitrogen in the operator was decreased by lowering the nitrogen line pressure to the valve from 50 to 25 psig.

The valve was subsequently tested and closed within the required time limit. The inspector had no further questions.

(2) On April 22, 1985, the inspector reviewed the documentation for maintenance on the containment high radiation monitor. This monitor was declared inoperable on April 16, 1985 and was repaired and subsequently returned to service on April 19, 1985.

No inadequacies were identified concerning the maintenance work. However, the inspector noted that the maintenance request (MR) No. 85-284, indicated that the equipment had been returned to service prior to a supervisory review of completed post work testing. The post work testing was completed on a Friday, April 19, 1985, but the testing was not reviewed until the following week.

The Chief Maintenance Engineer indicated that this incident was an isolated event, but that in the future, maintenance supervisors would be called into the plant on the weekends to review post work testing prior to closing out the maintenance requests. The inspector had no further questions.

(3) On April 19, 1985, the inspector discussed repairs to failed secondary containment isolation dampers AO-N-82 and 83. The repairs to these dampers are described in Section 4 of this report. The inspector had no further questions at this time.

8. Chemistry and Health Physics Activities

a. On April 16, 1985, the inspector attended a meeting of the radiological oversight committee. A variety of topics were discussed at the meeting, including recent housekeeping problems in plant and the use of radiological occurrence reports. A number of recommendations from the independent radiological assessor were also discussed.

While the discussions were open and forthright, they were protracted and few decisions were made. The inspector subsequently expressed concern about the committee's ability to review the large number of Radiological Improvement Program (RIP) milestones which will be completed in the next few months. The licensee indicated that the committee's efforts would become more efficient as it gains experience in its oversight function. The inspector had no further questions at this time.

b. The inspector reviewed the licensee's activities in preparation for a planned feedwater Hydrogen Water Chemistry (HWC) injection test at the site in order to determine the need for any additional shielding of plant equipment.

The inspector discussed system configuration and operation, and proposed radiological controls and surveys to be implemented during the test.

The licensee has utilized the services of Advanced Process Technology Co., and the General Electric Co. to assess the impact of HWC at the station, to assist in test implementation, and take measurements.

The inspector determined that the licensee had included an assessment of both onsite and offsite radiological impact (including comparison with regulatory dose rate restrictions). The licensee has also had special calibration studies done regarding survey instruments and monitoring devices response to the radionuclide involved. No problems were identified during this review.

c. The following information is included in this report to assist NRC management in following radiation exposure at the station. The monthly personnel radiation exposures for March and April, 1985 were 59.7 and 78.1 person-rems, respectively. The total yearly radiation exposure through May 5, 1985 was 266.5 person-rems.

The February monthly personnel radiation exposure was incorrectly reported in NRC Inspection Report 50-293/85-06. The reported exposure, 128.7 person-rems, was the yearly exposure through the end of February. The correct monthly exposure for February, 1985 was 52.2 person-rems. No inadequacies were identified.

9. Followup on Licensee Review of the Hatch Unit One Vent Header Cracking Incident

The licensee's initial response to the vent header cracking incident was reviewed during NRC Inspection No. 50-293/84-01. During the current inspection period, the licensee response to a General Electric Service Information Letter (SIL) No. 402, "Wetwell/Drywell Inerting", issued February 14, 1985 was reviewed. The Hatch event is attributed to the inadvertent injection of liquid nitrogen into a torus vent header.

The response to this SIL is documented in a letter from Boston Edison Co. to NRC Region I, dated September 14, 1984.

The inspector verified the accuracy and completeness of the response by discussing the SIL items with engineering and quality assurance department and by reviewing plant procedures and drawings.

No inadequacies in the licensee's actions were identified.

One potential source of liquid nitrogen which could affect the primary containment had not been fully evaluated at the time of the inspection. This source, the post-accident containment atmospheric dilution (CAD) system, is designed to deliver a relatively small volume (60 cfm) of gaseous nitrogen to the primary containment after an accident. The CAD system does not have low temperature cut off valves (present in other nitrogen lines) to prevent the accidental injection of liquid nitrogen into containment.

The licensee stated that the nitrogen supplied to the CAD system would normally be vaporized by truck-mounted vaporizers. The licensee plans to complete an evaluation of the effects of a failure of these vaporizers on the CAD system components and on the primary containment this summer.

The licensee evaluation of the CAD system will be reviewed during a future inspection (85-08-02).

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.

ATTACHMENT

The following is a list of surveillance and maintenance items reviewed during this period.

Portions of the following tests were reviewed:

-- Closure test of reactor water sample line valves on March 28, 1985.

-- Alternative system testing between April 2 and 3, 1985 due to an inoperative HPCI system.

-- Post maintenance full flow test on the HPCI system on April 2, 1985 (8.5.4.1).

-- Routine calibration of the refueling floor ventilation system radiation monitors on April 17, 1985 (6.5-170).

-- Routine functional test of the automatic initiation of the ADS on high drywell pressure (8.M.2-2.1.6) on April 24, 1985.

-- Routine calibration of reactor high pressure sensor PS-261-23A on April 22, 1985. Also reviewed calibration history from 1981 to April 1985.

Portions of the following maintenance items and temporary modifications were reviewed:

-- M.R. 84-1-55 Main steam line high flow switch replacement, environment qualification upgrade.

-- M.R. 85-284 Repair containment high radiation monitor.

-- M.R. 85-282 Repair damper AO-N-82.

-- M.R. 85-28 Repair damper AO-N-83.

-- M.R. 85-287 Repair flow compariter.

-- M.R. 85-267 Replace SV-220-44.

-- M.R. 85-45-107 APRM meter 3% low.

-- M.R. 85-289 AO-220-45 limit switch not working.

-- Temporary modification 85-22, remove demisters from the standby gas treatment system.

-- Temporary modification 85-25, AO-N-82 drive louver repair.