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U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-293/84-33

Docket No. 50-293

License No. DPR-35 Priority -- Category C

Licensee: Boston Edison Company

800 Boylston Street

Boston, Massachusetts 02199

Facility Name: Pilgrim Nuclear Power Station

Inspection Conducted: October 9, 1984 - November 19, 1984

Inspectors: M. McBride (for) 11/29/84
J. Johnson, Senior Resident Inspector Date

M. McBride 11/29/84
M. McBride, Resident Inspector Date

G. Meyer 12/6/84
G. Meyer, Project Engineer Date

Approved By: L. Tripp 12/6/84
L. Tripp, Chief, Reactor Projects Date
Section No. 3A, Projects Branch No. 3

Inspection Summary:
Inspection on October 9, 1984 - November 19, 1984 (Report No. 50-293/84-33)
Areas Inspected: Routine unannounced safety inspection of plant operations including followup of previous findings; operational safety verifications; followup on plant events and LERs; and a review of surveillance, maintenance activities, and refueling activities. The inspection involved 174.5 inspector-hours by two resident inspectors and one region-based inspector.

Results: No violations were identified. Concerns regarding concrete repairs to the reactor building truck lock are discussed in Details Section 3.b.(3).

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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. Followup on Previous Inspection Findings

   a. (Closed) Follow Item (83-19-04). Review over-current fault calculations for General Electric Co. metal clad switchgear. The licensee's calculations, No. E12-0 dated March 20, 1984, were reviewed by an NRC Region I specialist inspector and found to be acceptable to demonstrate that system modifications were not necessary. Worst case momentary short circuit currents do not exceed 60,000 amperes (compared to a design limit of 80,000 amperes). This item is closed.

   b. (Closed) Follow Item (84-04-03). Review cause and corrective actions for local leak rate failures of main steam isolation valves (MSIVs) and feedwater check valves. An NRC Region I specialist inspection was conducted (Report No. 84-31) to review the licensee's repair and modification activities regarding valve maintenance. This report describes the licensee's actions to correct the leak rate failures on the MSIVs and feedwater check valves as well as other containment isolation valves. This item is closed.

3. Operational Safety Verification

   a. Scope and Acceptance Criteria

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

   Observations included a review of equipment conditions, security, housekeeping, radiological controls, and equipment control (tagging). The inspectors also participated in an NRC team review of a remedial emergency preparedness exercise on November 8, 1984. The team review is documented in NRC Inspection Report No. 50-293/84-35.

b. Findings

(1) On October 11, 1984, the inspector discussed concerns with the licensee over the method of replacement of local power range monitor (LPRM) detectors in the reactor vessel. Blocks on the refueling bridge hoist which automatically stop hoist movement were being defeated to enable the new LPRM's to be installed. However, the blocks were not always reset prior to lifting irradiated LPRM's from the core. These blocks help prevent irradiated detectors from inadvertently being pulled to the surface of the pool, creating a substantial radiation hazard.

The licensee stated that personnel conducting the changeout had been counseled on the need to reset the blocks after each changeout and that the procedure for the replacement would be modified to require that the blocks be properly set. The inspector had no further questions. No violations were identified.

(2) The inspector reviewed the operational testing of control rod drives (CRD), a majority of which had been refurbished during the outage. The specific evolutions involved were venting of the CRD's, checks of coupling to the control rod, and timed travel checks. The testing and venting was performed to procedure No. 2.2.87, "Control Rod Drive System", September 26, 1984, which was reviewed by the inspector.

As part of the venting of the CRD's, the procedure specifies that while a CRD is withdrawing the control rod, a recorder trace be run of the differential pressure between drive water-in and drive water-out. This trace is used to confirm satisfactory venting. The inspector reviewed this phase of the testing at the control room, at the CRD hydraulic units (valving in and out of differential pressure transmitter), and at the recorder to verify that the testing was performed according to procedure by qualified personnel using properly calibrated testing equipment.

The CRD testing showed that problems existed with six CRD's. Specifically, CRD's 38-11 and 30-11 could not be withdrawn past position 46, CRD's 30-43 and 38-43 had excessive friction during travel beyond position 46, and CRD's 22-11 and 06-15 could not be coupled to the control rod. All of these problems appeared to relate to difficulties between the spud fingers of the CRD and the socket of the control rod. Subsequently, uncoupling of CRD 38-11 and underwater camera examination of the spud confirmed this; one spud finger was broken off, two spud fingers were bent, and the remaining three spud fingers were undamaged. The broken finger was retrieved by underwater vacuuming. The inspector observed the in-vessel work on CRD 38-11 and subsequently on CRD 30-11 to verify that procedures were followed and that proper radiological control practices were utilized.

The licensee decided to replace the spuds on the six problem CRD's. The inspector attended the pre-work briefing of the workers to verify that the work procedures and radiological control procedures were explained to the workers in sufficient detail prior to initiation of the work in the high radiation area.

Of the six CRD spuds replaced, three spuds were damaged (bent spud fingers). Subsequent to reinstallation the six CRD's were satisfactorily tested.

The inspector found the licensee's practices in this area to be acceptable and had no further questions.

(3) On November 7, 1984, the inspector noted that a QC Nonconformance Report (NCR) tag was attached to a repaired section of the concrete wall in the reactor building truck lock. A section of the wall had been taken down to facilitate the recirculation piping project and recently restored. This NCR (No. 84-211 dated November 2, 1984) stated that the concrete patch was not built in accordance with ACI 318 and Bechtel letter 10394BRME114, the required specification. Since core alterations were in progress, secondary containment integrity was required, and the inner truck lock door was open, the inspector questioned the licensee regarding the safety function of the reactor building truck lock. The licensee's QA Manager stated that a previously issued stop work order (No. 84-2 dated October 19, 1984) regarding the wall repairs was rescinded and an NCR tag was attached to assure that concerns (regarding rebar overlap, adequate curing, amount of water used, qualification of inspectors and records of inspection, void content, coating of forms, and aggregate size) were tracked for resolution. This change to an NCR was made after review by the AE, Bechtel Power Corporation, and the licensee's Nuclear Engineering Department (NED) Manager.

The inspector reviewed Bechtel letters No. 10394-BLE-3323 and 10394-BRME-114 dated October 25, 1984 and November 1, 1984 respectively. These letters state that the reactor building truck lock was not designed as a seismic class I structure, but that the wall area in question should be removed and reinstalled to the original repair scheme (ref. drawing SK-C-1431) after fuel loading in order to meet structural requirements for a class II structure. Further, NED Memo. No. 84-758 dated October 19, 1984 states that the reactor building truck lock is not and needs not be designed to be a class I structure.

The inspector held discussions with the NED Manager and his staff on November 8, 1984. The licensee explained further the design requirements of the reactor building truck lock and stated that the only safety function was to provide an air seal and that this was demonstrated by testing. The licensee stated that because it may be unclear, that the FSAR and list of safety related equipment (Q-List) would be revised. Pending further review of the design requirements of the reactor building truck lock, this item is unresolved (84-33- 01).

4. Followup on Events and Nonroutine Reports

   a. Reports

      (1) On October 9, 1984, visual indications of possible cracking were confirmed in the dry tube sections of a source range monitor and an intermediate range monitor. Indications were subsequently identified in a second intermediate range monitor dry tube. The indications were in the upper portions of the dry tubes, in sections which are not part of the primary coolant pressure boundary. The licensee issued a nonconformance report on the indications.

      The licensee stated that a General Electric safety evaluation indicated that the dry tubes can be used "as is" for another operating cycle. The safety evaluation has been accepted by the licensee's Engineering Group and forwarded to QA for review. The licensee plans to inspect and replace the affected dry tubes during the next refueling outage.

      Similar dry tube problems have been observed at other BWR's and are discussed in General Electric Company Service Information Letter (SIL) No. 409, dated June 19, 1984. No violations were identified.

      (2) On October 11 and November 3, 1984, automatic secondary containment isolation signals were generated when workers on the refueling floor of the reactor building inadvertently tripped refuel floor ventilation radiation monitors.

      The monitors, designed to detect elevated levels of radioactive material in the ventilation duct, were tripped when radioactive tools and equipment were brought too close to the monitors. In both cases, the reactor building isolated and the standby gas treatment system started. The licensee returned the system to a normal lineup after determining the cause of the isolation signals. The inspector had no further questions. No violations were identified.

      (3) On October 12, 1984, the licensee notified the NRC via the ENS telephone line that the turbine overspeed trip for the high pressure coolant injection system (HPCI) had been found broken during a once-per-cycle inspection of the HPCI turbine. The licensee stated that the trip mechanism had failed in the past and that it was scheduled to be replaced with a new type of trip prior to startup. This trip is not required to be operable by the Technical Specifications. The inspector had no further questions. No violations were identified.

(4) At 9:50 am on November 8, 1984, the control room was notified by an operator that a leak had developed in the control rod drive system down stream of the drive water filter. The operators immediately secured the control rod drive pumps and stopped all core alterations (refueling) while repairs were made. Maintenance Request No. 84-344 was approved for repair welding. The inspector discussed the event with the Watch Engineer on duty, reviewed system isolations, and reviewed drawing No. M250 sheet 2. The leak was in a non-safety related section of pipe where a pressure guage is welded to the two inch charging water header. The inspector verified that control rods and their associated accumulators were operable prior to the resumption of core alterations.

The licensee stated that the weld was not recently installed and may have been part of original plant construction. A circumferential hairline indication was identified during a penetrant examination of the weld. The weld was completely ground out and the pipe rewelded. No violations were identified.

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
84-04-01	Safety Valve Setpoints Below Requirement of Technical Specifications (update)
84-05-01	Target Rock Safety Relief Valve Operability Problems (update)
84-11	Unplanned start of the "A" Emergency Diesel Generator
84-13	Jet Pump Instrumentation Nozzle Indications
84-14	Unplanned Actuation of Engineered Safeguards Feature (Reactor Protection System)
84-15	Unplanned Actuation of Engineered Safeguards Feature (containment spray)

The problems with Target Rock safety relief valves reported in LER's 84-04-01 and 84-05-01 were reviewed during NRC inspections 50-293/84-23 and 84-31. Items discussed in LER's 84-13, 84-14, and 84-15 were reviewed during NRC inspection 50-293/84-26.

No violations were identified. However, the inspector discussed concerns with the licensee regarding the completeness and clarity of reports of events involving personnel errors. The licensee stated that efforts would be made to clarify future reports. The inspector had no further questions.

5. Surveillance Testing

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

Portions of the following tests were reviewed:

- Post maintenance testing of relief valves for the standby liquid control system

- Preparations for HPCI preoperational test, TP 83-69

- Post maintenance testing of scram pilot solenoid valves

- Once-per-day and once-per-shift refueling check lists

- Local leak rate testing of valve 2301-4 in accordance with procedure No. 8.7.1.5 on November 16, 1984

b. Findings

(1) On October 15, 1984, two relief valves on the standby liquid control system failed to lift during routine post work testing. The licensee stated that the valves had been rebuilt during the outage and set to lift at 1800 psig, the upper Technical Specification limit, which was above the post work test limit of 1425 $\pm$ 14 psig. The licensee removed and disassembled both valves and found no mechanical problems. The valves were reinstalled and functioned properly. No violations were identified.

(2) On October 15, 1984, the licensee received a spurious scram signal from the high water level sensors in the west control rod drive scram discharge volume (SDV) during testing on the east SDV. The licensee determined that small amounts of water running down the interior sides of the west SDV were splashing onto the high water level sensors, generating a spurious trip signal. The licensee installed splash guards on the sensors and has had no further problems. No violations were identified.

(3) On October 25, 1984, the inspector reviewed the preparations for flushing of sections of the HPCI coolant piping, modified under Plant Design Change 81-38. The flushing was to be performed under preoperation test procedure (TP) 83-69, HPCI Coolant Loop Flush and Hydro, October 19, 1984, implemented

by Maintenance Request (MR) 84-23-103, October 22, 1984 and by RWP 84-2802. The inspector reviewed the above administrative controls for the testing and found them to be acceptable. The inspector walked down the modified HPCI piping to verify that the installed configuration presented no problems or concerns that were not already covered under the test procedure. The inspector found no such concerns.

(4) During a control room tour on October 25, 1984, the inspector reviewed the testing equipment which had been used as part of the calibration of source range and intermediate range monitoring equipment. The inspector found that megohmmeter P-16 had exceeded its calibration due date (October 12, 1984). In later discussions, instrument and controls (I&C) personnel stated that the megohmmeter had not been used to make any detailed measurements required by the calibration procedure, but had only been used to aid the technician in troubleshooting any test setup problems. The inspector stated that although no regulatory requirements were violated, the field use of any test equipment not properly calibrated represented a poor practice. The licensee representative agreed to remove the megohmmeter and replace it with one properly calibrated and to reinstruct I&C personnel to use only equipment within their calibration dates. The inspector had no further questions.

(5) On October 30, 1984, the inspector reviewed post work testing on scram pilot valves. This item is discussed in Section 6 of this report.

(6) The inspector reviewed a sampling of refueling checklists during the period October 31 - November 16, 1984. No violations were identified.

6. 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.

The items and documents reviewed included the following:

- M.R. 83-3-14, Replace BUNA-N material in scram pilot solenoid valves for control rod drive hydraulic units

b. Findings

(1) On October 12, 1984, the inspector reviewed the use of a non-Q feeder breaker (B602) in a safety-related 480 V bus (B6). The breaker is one of two feeder breakers on B6 and was temporarily installed during a breaker overhaul program. The bus was not required to be operable at the time (i.e. during the outage prior to core reload) but was energized. The temporary breaker was similar, but not identical to the normally installed breaker. The licensee controlled the installation and removal of the temporary breaker with a maintenance request and testing tags. The inspector verified the temporary breaker had been removed for reactor refueling. No violations were identified.

(2) On October 16, 1984, the licensee completed an evaluation of potential environmental qualification problems with Rosemont model 1153, series "B" transmitters. These transmitters are installed in the scram discharge volume high level trip system and is part of the reactor protection system.

The licensee received a letter from Rosemont, dated September 10, 1984, which indicated that a potential leakage path into the transmitter housing had been identified which could affect the transmitters' stability in a moist environment. The letter was entitled, "Notification under 10 CFR Part 21 Regulations".

The licensee stated that the purchase documents specified that the transmitter be environmentally qualified and was subject to the reporting requirements of 10 CFR 21. However, a licensee review indicated that the environmental qualification requirement was too conservative since the transmitters did not need to function after a plant scram. As a result, the licensee does not believe that the transmitters are unacceptable and does not plan to report the problem to the NRC under 10 CFR 21.

The licensee stated that the vendor (Rosemont) had been contacted and had stated that the NRC had not been directly notified under 10 CFR 21. Rosemont instead had notified individual licensee's of the problem. No violations were identified.

(3) On October 18, 1984, the inspector reviewed the status and condition of ELMA Engineering power supplies which were purchased by Nutherm International Inc., provided to Boston Edison Co., and installed in the scram discharge volume reactor protection system analog trip units. The inspector discussed with the licensee's Senior I&C Engineer a 10 CFR 21 report that had been issued by another nuclear facility regarding defective wiring, cold solder joints and overall poor workmanship in ELMA power supply units. The licensee had been aware of the need to change out capacitors and had done so. They also had replaced one of the eight new power supplies because of failure to meet voltage regulation requirements upon installation. The failed power supply was inspected by the licensee and observed by the inspector in the I&C lab.

The licensee did not identify any other rejectable deficiencies because of cold solder joints and poor workmanship, acknowledged the information regarding problems at another facility and stated that continued monitoring would be performed to ensure future reliability. The inspector had no further questions or concerns at this time.

(4) On October 19, 1984, the inspector reviewed a procedure that was written by the Bechtel Power Corporation maintenance staff for core drilling into concrete structures and approved by the licensee's Nuclear Engineering Department, Construction Management Group (CMG) Leader. The inspector expressed concern to the licensee's station management personnel regarding the requirements of the On-site Review Committee to review and approve a procedure of this type although it had not been used. The Station Manager acknowledged the inspector's concerns, and initiated actions to implement the proper review and approval process. The inspector had no further questions at this time.

(5) On October 30, 1984, the inspector reviewed the licensee's scram pilot valve maintenance program and discussed recent events regarding defective polyurethane disc materials at another nuclear facility.

The licensee has implemented a preventative maintenance program in accordance with G.E. SIL No. 128 regarding periodic in-kind replacement of BUNA-N parts in the solenoid valves. This rebuilding of the pilot head and body assemblies is described by procedure No. 3.M.2-16 and was implemented by Maintenance Request (M.R.) No. 83-3-14 between April and May, 1984 for all 145 control rod drives. The licensee's QC group reviewed this maintenance request and performed inspections during the rebuilding.

The inspector had no further questions regarding the material replacement but noted that there was no post maintenance operability test documented on the M.R. Following discussions with the inspector, the licensee revised procedure No. 8.M.1-22, and conducted a functional test via a reactor scram signal. This test was performed at 8:00 pm on October 30, 1984 and demonstrated post maintenance operability of the scram pilot solenoid valves.

The inspector had no further questions. No violations were identified.

7. Refueling Activities

a. Scope

The inspector reviewed the licensee's actions associated with reinstallation of the fuel assemblies from the spent fuel pool into the core following the recirculation piping replacement. This review included preparations and prerequisites for core alterations, witnessing of fuel movement from the refueling bridge, and control room observations. These items were reviewed in order to verify proper implementation of the requirements of the Technical Specifications and the licensee's procedures.

b. Findings

(1) The inspector reviewed and verified that the pre-refueling check list (OPER 10) had been completed at 12:30 pm on October 30, 1984. This included verification that the On-site Review Committee had granted permission to move fuel. The inspector also reviewed secondary containment test data sheets and verified that the tests performed on October 28, 1984 demonstrated acceptable performance of building integrity and the standby gas treatment system. The inspector also independently verified operability of the refueling floor ventilation process radiation monitors and completion of surveillance testing for the emergency diesel generators, the core spray system, the refueling platform and reactor manual control system. No violations were identified.

(2) On November 1, 1984, the licensee was not aware that the "B" source range monitor (SRM) was bypassed during refueling activities. The licensee subsequently determined that fuel and control rods located in the "B" SRM quadrant had been moved while the SRM was bypassed. On November 7, 1984, the licensee failed to continuously monitor the SRM's while a fuel assembly was loaded into the reactor. These items were reviewed during an NRC Special Inspection and documented in Inspection Report 50-293/84-36.

(3) On November 5, 1984, the inspector observed fuel movements from the refueling bridge. A Senior Reactor Operator supervised the refueling bridge activities and was assisted by a licensed Reactor Operator. The activities were well controlled and in conformance with the requirements of the Technical Specifications and procedure No. 4.3, "Fuel Handling." No violations were identified.

8. Personnel Radiation Exposure Tracking

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

During the month of October, 1984, 345 person-rems (based on TLD measurements) of radiation exposure were received from outage activities. The September exposure was less than the previous monthly exposure, 374 person-rems, reflecting a decrease in the number of man-hours spent on outage tasks in October. Approximately 30% of the exposure was from recirculation project work, about 20% from control rod drive work, and the remainder from other outage tasks.

The total outage exposure at the end of October, 1984, was about 3,700 person-rems (based on TLD measurements) which is equivalent to 92% of the projected exposure through the end of October. About 1,500 person-rems (or 40%) of the outage exposure has been attributed to the recirculation project. The licensee is considering adjusting the recirculation project exposure upwards and the nonrecirculation exposure downwards by as much as 350 person-rems to more accurately reflect health physics personnel exposure and non-radiation work permit exposure.

9. 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.