U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 50-293/90-19

Licensee: Boston Edison Company (BECo)
800 Boylston Street
Boston, Massachusetts 02199

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: July 24 - August 15, 1990

Inspectors: J. Macdonald, Senior Resident Inspector
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Approved by: John T Rogge 9/17/90
J. Rogge, Chief, Reactor Projects Section 3A Date

Inspection Summary: Inspection on July 24 - August 15, 1990 (Report No. 50-293/90-19)

Areas Inspected: Routine safety inspection of actions on previous inspection findings, operational safety, plant operations, security, maintenance and surveillance, engineering support, radiological controls, safety assessment/quality verification and periodic reports.

Results: Inspection results are summarized in the attached Executive Summary. No violations or unresolved items were identified during this inspection period.

EXECUTIVE SUMMARY

Pilgrim Inspection Report 50-293/90-19

July 24 - August 15, 1990

Operations: Operator response to the July 26 turbine stop valve closure transient was excellent (Section 2.5.1). Conservative actions were taken by control room personnel during the July 24 electrical storm and resultant loss of an offsite transmission line, with noteworthy command and control demonstrated by the Nuclear Watch Engineer (section 2.5.2).

Maintenance/Surveillance: Good communications between maintenance department and systems engineering personnel were evident during the turbine stop valve repair effort and during the evaluation and replacement of a failed main steam line pressure transmitter.

Security: The security program continued to be properly implemented. BECo's request for a management meeting to discuss actions taken in response to a recent security inspection was a good initiative.

Engineering/Technical Support: Salt service water (SSW) pump discharge column replacement continued in a controlled and deliberate manner.

Safety Assessment and Quality Verification: Operational Review Committee review of LER 90-10 pertaining to the July 3 reactor shutdown was extensive and greatly improved the technical quality of the report.

Radiological Controls: Station endorsement of the ALARA program was evidenced by the successful execution of exposure controls during the spring outage.

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*The NRC Inspection Manual inspection procedure (IP) that was used as inspection guidance is listed for each applicable report section.

DETAILS

1.0 Summary of Facility Activities

Pilgrim Nuclear Power Station (Pilgrim, PNPS, or the licensee) was at full power at the start of this report period. On July 26, while reducing reactor power to maintain condenser back pressure, the Number 2 Turbine Stop Valve (section 2.5.1) unexpectedly closed; the cognizant reactor operator immediately reduced power to 70%, thereby precluding a reactor scram. Power was further reduced to approximately 45% on July 28 in order to perform testing of the "A" Recirculation Pump Motor-Generator set. On July 29, BECo commenced a further power reduction to facilitate removal of the turbine-generator from the grid in order to perform latch mechanism repair to the Number 2 Turbine Stop Valve. On July 30, at 3:53 a.m., the turbine-generator was synchronized to the grid. The plant reached full power on August 5 at 7:05 a.m. At the conclusion of this report period, the plant was at full power.

On July 24, BECo notified the NRC Operations Center via the Emergency Notification System (ENS) of a spurious closure of three hydrogen/oxygen sampling valves during an electrical storm, possibly due to a momentary lightning induced voltage transient (section 3.2). This notification was made in accordance with 10 CFR 50.72. Notification via ENS to the NRC was also made on July 31 after the licensee notified the Town of Plymouth Fire Department of a potential diesel fuel oil leak from the station blackout diesel underground double-lined storage tank.

On August 13, the Director of Project Directorate I-3 from the Office of Nuclear Reactor Regulation (NRR) responsible for Pilgrim oversight, was onsite to meet with the NRC resident inspectors. On August 16, the Director of the Office of NRR was onsite to meet with the NRC resident inspectors.

The following NRC Region I specialist inspection was conducted during this report period:

a. Engineering and Technical Support, August 6-10, 1990 (Inspection Report 50-293/90-09)

2.0 Operational Safety

2.1 Plant Operations Review

The inspectors observed, during regular and backshift hours, the following areas:

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

Control room instruments were observed for correlation between channels, proper functioning and conformance with Technical Specifications. Alarms received in the control room were reviewed and discussed with the operators. Operator awareness and response to these conditions were reviewed. Operators were found cognizant of board and plant conditions. Control room and shift manning were compared with Technical Specification requirements. Posting and control of radiation, contaminated, and high radiation areas were inspected. Use of and compliance with radiation work permits and use of required personnel monitoring devices were checked. Plant housekeeping controls, including control of flammable and other hazardous materials, were observed. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were corrcctly made and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout and lifted lead and jumper logs. Inspections were performed on backshifts including July 26 and 31, August 1, 2, 3, 7, and 14.

Pre-evolution briefings were thorough. The operators demonstrated good knowledge of plant conditions. No unauthorized reading material was observed and food, beverages and hard hats were kept away from control panels.

2.2 Safety System Review

Portions of the emergency diesel generators, reactor core isolation cooling, core spray, high pressure coolant injection, residual heat removal and safety related electrical systems were reviewed to verify proper alignment and operational status in the standby mode. The review included verification that (i) accessible major flow path valves were correctly positioned, (ii) power supplies were energized, (iii) lubrication and component cooling was proper, and (iv) components were operable based on a visual inspection of equipment for leakage and general conditions. No violations or safety concerns were identified.

2.3 Operational Safety Findings

Licensee administrative control of off-normal system configurations by the use of temporary modifications and tagging procedures was in compliance with procedural instructions and was consistent with plant safety. Backshift inspections found operators to be alert and attentive. Overall plant cleanliness and material condition continued to be good. However, isolated instances of loose debris were observed during routine plant tours. These conditions were quickly remedied when brought to the attention of BECo management.

2.4 Inoperable Equipment

Actions taken by plant personnel during periods when equipment was inoperable were reviewed to verify that: technical specification limits were met, alternate surveillance testing was completed satisfactorily, and equipment was properly returned to service upon completion of repairs. This review was completed for the following items:

Date Out	Date In	System
7/24	7/25	"A" Salt Service Water Pump
7/26	7/26	Augmented Offgas System Radiation Monitor
7/28	7/28	"A" Recirculation Pump Motor-Generator Set
7/29	7/29	"A" Recirculation Pump Motor-Generator Set
8/1	8/2	Station Blackout Diesel-Generator

2.5 Plant Operations

2.5.1 Turbine Stop Valve Closure and Transient

On July 26 at 10:30 p.m., while the licensee was reducing power in order to maintain condenser vacuum, the Number 2 Turbine Stop Valve unexpectedly went closed. The control valves responded as expected by opening further. In response, the reactor operator quickly reduced recirculation flow to run back reactor power to 70%. This effectively limited the reactor pressure increase to only 10 psig, from 1036 psig before the transient to a maximum of 1046 psig during the transient. All other plant responses were as expected.

The reactor operator and control room personnel demonstrated excellent control of the transient, as noted by inspector review of the EPIC computer traces and printout. Quick operator runback of recirculation flow combined with the slow closure of the turbine stop valve limited the pressure transient and prevented a reactor scram.

Licensee investigation determined that the stop valve bypass control mechanism trigger finger on the latch assembly was slightly rounded due to normal wear. The trigger finger on the latch assembly was repaired by adding several weld beads on the latch surface interface and ensuring a sharp edge. Excellent follow-up and troubleshooting by the systems engineer was considered a strength.

2.5.2 Partial Group II PCIS Isolation

On July 24 at 2:12 p.m. during an electrical storm, air circuit breakers (ACBs) 103 and 104 in the switchyard tripped open. ACB 103 reclosed automatically. ACB 104 was manually closed 23 minutes later, after BECo determined that no damage had occurred. However, after the trip of ACBs 103 and 104, control room operators noticed that three Group II (hydrogen/oxygen sampling system) isolation valves had closed unexpectedly.

The three sampling system isolation valves which closed were SV-5065-20B, SV-5065-21A and SV-5065-14A. Licensee investigation of the closure of the valves was still ongoing at the conclusion of this report period. However, BECo believed that the partial actuation of the Primary Containment Isolation System (PCIS) relays was caused by a momentary (20 to 40 millisecond) transient voltage decrease due to the electrical storm. The voltage decrease was apparently sufficient to de-energize the 16A-K17X relay and resulted in the isolation of the three sampling system valves. The fourth sampling system valve, which remained open, was powered from a different power supply.

The sampling system isolation valve closure was similar to a previous event. On May 3, 1989 following an automatic turbine trip, reactor scram and resultant fast transfer of the 4160V power sources, the same three sampling valves isolated (see Inspection Report 50-293/89-06, section 2.3.3). The details of this event, root cause and corrective actions were delineated in Licensee Event Report 89-015-00. The licensee considered the two events similar in that a momentary transient voltage decrease de-energized a CR120 relay and caused the valves to isolate.

Actions by the control room staff in response to this event were prompt and conservative. During the storm, the control room panels were regularly walked down to determine if any additional abnormalities or spurious isolations had occurred. Command and control by the Nuclear Watch Engineer (NWE) was excellent. The inspector will monitor licensee actions in this area during routine inspection efforts.

3.0 Maintenance/Surveillance

3.1 Replacement of a Main Steam Line Pressure Transmitter

On August 1, a Main Steam Line Channel B low pressure alarm and a half permissive trip condition was received for the main steam isolation valves (MSIVs). The EPIC (Emergency and Plant Information Computer) trace for the affected channel indicated a false pressure step decrease to about 763 psig, low enough to cause the control room alarm. After several spikes, the indicated pressure returned to about normal (about 980 psig) and stabilized. Failure and malfunction report (F&MR 90-247) and maintenance request 90-45-283 were written for troubleshooting the cause of the alarm.

The erratic main steam line pressure transmitter, PT-261-30B, was replaced with a new transmitter. The failed pressure transmitter was a Rosemount Model 1153 transmitter. Initial licensee analysis of the event concluded the transmitter exhibited the oil impurity symptoms previously identified by Rosemount as a generic concern. However, the failed transmitter was believed to have been manufactured since Rosemount took correct actions to resolve these oil impurity problems. The transmitter was not manufactured from a suspect lot identified in NRC Bulletin 90-01. The transmitter was installed in September 1989 and had not shown signs of degradation or symptoms of possible failure prior to this event. The transmitter was returned to Rosemount for root cause analysis. The inspectors had no further questions.

4.0 Security

4.1 Observations of Physical Security

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify that controls were in accordance with the security plan and approved procedures. This review included the following security measures: security officer staffing, vital and protected area barrier integrity, and implementation of access controls including access authorization and badge issue, searches of personnel, packages and vehicles and escorting. No inadequacies were identified.

On July 31 a meeting was conducted in the Region I office between selected BECo and NRC security managers. BECo requested the meeting to discuss actions implemented or planned in response to the recent security specialist inspection (50-293/90-16).

5.0 Engineering/Technical Support

5.1 Salt Service Water Pump Modification Update

On January 11, 1990 while transporting an assembled Salt Service Water (SSW) pump from the maintenance shop to the intake structure for reinstallation, the bowl/impeller end of the "A" SSW pump was dropped approximately 6 to 12 inches. This led to a 360 degree through-wall brittle fracture of the second, 80 inch column section from the bowl/impeller end (see Inspection Reports 50-293/90-05, section 6.2 and 50-293/90-07, Attachment III).

In response to the column fracture, BECo conducted extensive metallurgical analysis of the failed column as well as the remaining SSW pump columns to determine material properties. Although only the failed column yielded unsatisfactory material characteristics, BECo ultimately elected to replace all SSW pump columns. To date, the columns on the "B", "C" and "D" SSW pumps have been replaced with new heat treated nickel aluminum bronze centrifugally cast columns. The presence of nickel stabilizes the aluminum bronze alloy and reduces potential eutectoid formations, and improved heat treatment processes enhance corrosion resistance. The "A" and "E" SSW pump columns are scheduled to be replaced with the new columns by mid-October. The two pumps have a mixture of both old and new columns; tie rods remain installed on these pump columns.

Concurrent maintenance activities being performed on the salt service water pumps include: (1) overhaul of the SSW pump impeller ends to ensure maximum useful life; (2) research of new methods to reassemble the pumps horizontally; (3) provisions to maintain an assembled spare pump available for installation, (4) manufacture of new SSW pump motor support assemblies (transition pieces), and (5) installation of stabilizing legs on all pumps to reduce vibration, along with investigation of alternate design methods to also reduce vibration. The inspectors will continue to monitor this area during routine inspections.

6.0 Safety Assessment/Quality Verification

6.1 (Closed) Violation 90-05-01 and Unresolved Item 90-05-02, Reactor Coolant System (RCS) Instrument Line Flow Check Valve Operability Issues

On February 9, 1990, BECo determined the operability of two RCS (Reactor Coolant System) instrument line flow check valves had not been properly verified as required by TSs (Technical Specifications). They subsequently declared the valves inoperable and entered the appropriate 24 hour LCO. This issue was documented in NRC Inspection Reports 50-293/89-12 and 90-05.

BECo immediately requested and was granted an NRC temporary waiver of compliance from the requirements of the instrument line flow check valve TS until the March 1990 outage. Additionally, BECo instituted administrative and physical controls to ensure degradation of either valve would be expeditiously identified and isolated. Long-term corrective actions detailed in the licensee response (dated May 1, 1990) to the Notice of Violation included: replacement of the affected valves with valves capable of being tested in place (accomplished during the March outage; see NRC IR 50-293/90-07); review of recently completed surveillance procedures to ensure similar problems were not present elsewhere; conduct of a human performance evaluation system investigation of the entire scenario, and a review of the modification, surveillance, F&MR, and TS clarification processes to determine if potential generic weaknesses were identified by this issue.

On March 29, 1990, a management meeting was conducted in the Region I office to discuss the implementation of BECo's corrective actions to this issue. The licensee's presentation was comprehensive and indicated extensive evaluation of the issue. The corrective actions implemented by the licensee properly addressed this issue and appear appropriate to preclude recurrence. The inspector had no further questions. These items are closed.

6.2 (Closed) Unresolved Item 87-53-02.5, Strengthen Communications Practices to Assure Clear Understanding and Directed Actions

The Augmented Inspection Team (AIT) Report Item 50-293/87-53-02.5 noted several planning weaknesses exhibited by BECo. One planning weakness contributed to poor communications between the operations and maintenance organizations. In addition, control and communications weaknesses were involved in the shutdown of the "B" emergency diesel generator.

Since 1987, BECo has demonstrated continuing improvement in the area of communications throughout the plant. Pre-evolution briefings are routinely conducted to ensure all personnel involved in a job are cognizant of job activities and their specific task. This includes the performance of surveillances, where pre-evolution briefs have been noted to be thorough and concise. Communications during performance of both phases of the Shutdown Outside the Control Room Test were noted to be excellent. During emergency preparedness drills and exercises, good information flow between the control room and Technical Support Center was noted to be an exercise strength. Another example of effective internal and external communications was prior to and during the declaration of an Unusual Event (NOUE) on July 3 (Inspection Report 50-293/90-15, section 9.1). Routine control room communications have also been noted by the inspectors to be effective and professional. Based on the above, this item is closed.

6.3 Closure of Outstanding NUREG-0737

The following NUREG 0737 items have been addressed in various correspondence between the licensee and the NRC. The inspector reviewed the following items and determined they were appropriately implemented as follows.

6.3.1 (Closed) Item I.A.1.3.1, Shift Manning - Limit Overtime

The guidelines for approval of overtime for licensee personnel who perform safety-related work contained in Generic Letter No. 82-12 were incorporated into licensee procedures in

1984. The inspector reviewed the BECo Nuclear Organization Mission, Organization and Policy Manual Section C.4.05, "Overtime Control," (Revision 1 dated 5/22/90) and operating procedure 1.3.67, "Use and Control of Overtime at PNPS," (Revision 2 dated 3/24/89) and determined GL 82-12 overtime requirements were properly implemented. Additionally, the inspectors review overtime records as a course of routine inspections. Most recently, licensed operator overtime was reviewed in NRC Inspection Report 50-293/90-13; no problems were noted. This item is closed.

6.3.2 <u>(Closed) Item II.D.3.1, Valve Position Indication - Install Direct Indications of Valve Positions</u>

The licensee installed a non-safety-related, single channel, seismically qualified accoustic monitoring package (Technology for Energy Company) with control room indication for the four safety relief valves and the two code safety valves. A plant design change (PDC 90-04), is being developed to incorporate current Regulatory Guide 1.97 criteria, which previously did not exist and were not a design requirement of the TMI Action Plan, into the current design. The inspector will review the PDC implementation in a future inspection report. This item is closed.

6.3.3 <u>(Closed) Item II.E.4.2.7, Containment Isolation Dependability - Radiation Signal on Purge Valves</u>

The Boiling Water Reactor Owners Group (BWROG) provided the NRC a generic evaluation which concluded that an automatic high radiation isolation signal for containment purge and vent valves was not necessary. The NRC staff issued a generic safety evaluation to the BWROG (dated May 7, 1986) accepting the BWROG position for purge and vent valves two inches or smaller in diameter. By letter dated March 14, 1988, as supplemented on March 27, 1989 and June 30, 1989, BECo requested that the requirements of Item II.E.4.2.7 not be extended to two-inch or smaller containment purge and vent valves. The request was based on a site specific application of the BWROG evaluation. On August 16, 1989, the NRC staff concluded that the licensee demonstrated the applicability of the BWROG evaluation to PNPS.
This item is closed.

6.3.4 <u>(Closed) Item II.F.2.4, Instrumentation for Detection of Inadequate Core Cooling</u>

During refueling outage RFO7, BECo implemented plant design change PDC 85-07, Reactor Water Level Instrumentation Modification. The design change incorporated the recommendations of NRC Generic Letter 84-23, Reactor Vessel Water Level Instrumentation in BWRs. The reactor water level instrumentation reference legs were relocated to a location outside the drywell and the vertical drop of the reference legs was reduced to approximately one foot. Additionally, the mechanical level indication equipment (Yarway) was replaced with an analog trip system. On February 27, 1986, as corrected on October 20, 1986, the licensee submitted a proposed TS change to address the requirements for instrument channel test and calibration frequencies. On March 3, 1987, the NRC issued TS amendment No. 99 to the PNPS Technical Specifications, authorizing the proposed changes. The implementation of PDC 85-07 provided improvements in the accuracy and reliability of the reactor water level monitoring control system and core cooling instrumentation, thereby ensuring initiation of safety system actuations independent of drywell conditions. This item is closed.

6.3.5 (Closed) Item II.K.3.18.C, ADS Modification

This item was previously addressed and closed in NRC Inspection Report 50-293/90-03, section 3.1, but was not removed from appropriate tracking systems. This item remains closed.

6.3.6 (Closed) Item II.K.3.27, Common Reference Levels for BWRs

This item required that a common zero reference be established for reactor water level measurement. During RFO7, the licensee implemented PDC 84-70, which established the common zero reference for reactor water level instrumentation as the bottom of the steam separators which are 482.5 inches above the bottom of the vessel and are 127.5 inches above the top of active fuel. Prior to the design change, several zero reference points existed; vessel level instrumentation and temporary templates were used to standardize the point of reference.

The inspector reviewed portions of the design change, affected procedure and design bases documentation revisions and had no questions. This item is closed.

6.3.7 (Closed) Item II.K.3.57, Identification of Water Sources Prior to Manual Activation of ADS

Inspector review of the licensee emergency operating procedures (EOPs) determined that throughout the flowpath formatted procedures (consistent with Rev. 4 EOP guidelines) a minimum reactor vessel water level of -126.5 inches (height of the top of active fuel) was referenced as being necessary to manually initiate ADS or continue automatic initiation sequences. Additionally, the NRC EOP team inspection (50-293/88-11) determined that the licensee EOPs were adequate. The inspectors had no further questions. This item is closed.

7.0 Radiological Controls

7.1 ALARA Reports

The inspectors reviewed the following ALARA reports:

-- ALARA Committee Meeting Minutes (7/17 & 7/24) dated July 26, 1990
-- 1990 Mid-Cycle Outage ALARA Report, dated August 8, 1990

The spring mid-cycle outage, which was extended nine days as a result of emergent work, was conducted from March 12 through April 28, 1990. Total radiation exposure for the outage was 67.98 Rem, which was 12% above the projected exposure of 60 Rem. The excess was largely attributed to unplanned emergent work throughout the outage as well as the nine day outage extension. ALARA outage planning was developed over a three month period, with the scope being frozen on February 15, 1990. An ALARA specialist was dedicated to the review of each outage task. Additionally, the ALARA review process was augmented by the temporary assignment of two radiological protection engineers and the maintenance section radiological advisor.

The inspector's review of individual outage task exposures indicated that actual exposure for planned activities in which ALARA reviews and dose reduction measures were implemented was approximately 25% less than estimated. The exposure savings were largely the result of training which utilized mockups, pre-staging of equipment, temporary shielding, implementation of contamination control measures, and incorporation of worker suggestions for improvement into work packages. Additionally, the ALARA job history file, initiated in 1986, established a reference for the incorporation of lessons learned into current work activities and provided additional opportunity for exposure reductions. Specifically, the application of lessons learned from maintenance on the "B" regenerative heat exchanger, as well as application of the above exposure savings techniques, enabled the same maintenance to be performed on the "A" regenerative heat exchanger with an approximate savings of 40 man-hours and 4.8 Rem. Conversely, emergent work activities in which minimal ALARA planning was possible resulted in actual radiation exposure being more than 20% greater than estimated for these tasks.

Contamination control measures were also effective during the outage. BECo utilized catch containments, equipment laydown areas, HEPA vents and vacuums, temporary tents and containments, and glove boxes to control contaminated material. Five personnel contaminations were reported during the outage. All were less than 10K dpm. No personnel contaminations resulted from drywell work activities.

In conclusion, the ALARA program continued to be well supported by plant management and ALARA awareness continues to be a focal point of plant staff. The Spring 1990 outage ALARA planning was effective in minimizing personnel exposures and contamination controls were effective in minimizing personnel contamination events.

8.0 Review of Periodic and Special Reports

Upon receipt, the inspectors reviewed periodic and special reports submitted pursuant to the Technical Specifications. This review verified, as applicable: (1) that the reported information was valid and included the data required by the NRC; (2) that test results and supporting information were consistent with design predictions and performance specifications; and (3) that planned corrective actions were adequate for resolution of the problem. The inspectors also ascertained whether any reported information should be classified as an abnormal occurrence. The following reports were reviewed:

-- Monthly Operational Status Summary for July 1990
-- Operations Review Committee and Nuclear Safety Review and Audit Committee Meeting Minutes
-- 1990 Spring Outage ALARA Report
-- ALARA Committee meeting minutes

9.0 Management Meetings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss licensee activities and areas of concern to the inspectors. On August 24, the resident inspector staff conducted an exit meeting with BECo management summarizing inspection activity and findings for this report period. No proprietary information was identified as being included in the report.