U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 50-293/92-21

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

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: September 8 - October 19, 1992

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Approved by: ______________________ for   11/6/92
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Scope: Resident safety inspections in the areas of plant operations, radiological controls, maintenance and surveillance, emergency preparedness, security, safety assessment and quality verification, and engineering and technical support. Initiatives selected for inspection included coordination of maintenance, implementation of a design modification for improved electrical connectors, and operability determination training (NRC Generic Letter 91-18).

Inspections were performed on backshifts during September 11, 14-17, 21-25, 28-30 and October 1, 6-8, and 13-15. "Deep backshift" (10:00 p.m. - 5:00 a.m.) inspection was performed on September 24 and 25, and October 8 and 9.

Findings: Violation 90-25-01 concerning adherence to radiation protection procedures was reviewed and closed. Inspection results are summarized in the Executive Summary.

EXECUTIVE SUMMARY

Pilgrim Inspection Report 50-293/92-21

Plant Operations

Control room operators maintained excellent awareness of maintenance conducted during thermal backwash of the main condenser. Effective interdepartmental communications and coordination were evident. Proper safety perspectives and knowledge of licensing bases were demonstrated during evaluation of the hydrogen oxygen analyzer.

Radiological Controls

Management continued to support initiatives to minimize personnel radiation exposure. Increased awareness of radiation protection requirements, excellent training, and thorough management tours of the RCA were observed. A comprehensive, centralized instruction was instituted to better define radiation work practices and station program expectations.

Maintenance and Surveillance

Maintenance Department personnel effectively completed several corrective maintenance activities during the reactor downpower to conduct a main condenser backwash. Technician mockup training provided component familiarity and procedural critique that, in turn, enhanced recirculation pump motor generator set brush replacement. Good planning and excellent communication with control room personnel were evident. "Quick-connect" electrical connectors were installed to address potential age related fatigue of area temperature switch terminal blocks, resulting in increased RCIC system reliability.

Emergency Preparedness

Drills simulating hazardous material and an injured contaminated worker provided good organizational response training that included interactions with offsite support agencies.

Security

Continued alert security force performance was observed, both during regular hours and backshift inspections.

Safety Assessment and Quality Verification

Implementation training modules for the 50.59 safety evaluation program were comprehensive and procedural instruction was adequate. Licensee Event Reports (LERs) accurately described issues and appropriately addressed reporting criteria. The steam leakage path from Target Rock safety relief valves, as described in LER 92-08, warrants further consideration relative to future evaluations of leakage.

(EXECUTIVE SUMMARY CONTINUED)

Engineering and Technical Support

Erection of temporary scaffolding in preparation for mid-cycle outage activities was emphasized at the plan-of-the-day meeting and controlled in accordance with established procedures, with one minor exception (regarding seismic qualification) that was promptly corrected. Training associated with operability decision-making for degraded or nonconforming conditions was consistent with generic NRC guidance.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

At the start of the report period Pilgrim Nuclear Power Station (PNPS or Pilgrim) was operating at approximately 100% of rated power. On September 12, the "A" drywell hydrogen/oxygen analyzer failed a monthly functional test and was declared inoperable. The hydrogen portion of the analyzer was declared operable on September 16, 1992 (Section 2.2).

On September 13, reactor power was reduced to approximately 50 percent for main condenser backwash and several maintenance activities which included repack of the "A" condensate pump and replacement of the "B" recirculation pump motor generator set exciter end brushes. Reactor power was returned to 100 percent on September 15 following successful completion of all planned maintenance with the exception of the "A" condensate pump. The pump continued to run with significant seal leakage (Section 4.1) until repairs were performed on September 28.

On September 21 and 24, the reactor water cleanup and reactor core isolation cooling systems respectively were declared inoperable for brief periods of time to support installation of an electrical connector design modification (Section 4.2). On September 28, reactor power was reduced to approximately 75 percent to facilitate repairs to the "A" condensate pump seal. A temporary clamp was installed and Furmanite injected to successfully repair the "A" condensate pump seal. The reactor was returned to full power which was maintained through the end of the reporting period.

2.0 PLANT OPERATIONS (71707, 40500, 90712)

2.1 Plant Operations Review

The inspectors observed plant operations during regular and backshift hours in the following areas: Control Room, Protected Area Fence Line, Reactor Building, Diesel Generator Building, Turbine Building, Switchgear Rooms, Screen House, and Security Facilities

Control room instruments were independently observed by NRC inspectors and found to be in correlation amongst channels, properly functioning and in conformance with Technical Specifications. Alarms received in the control room were reviewed and discussed with the operators. Operators were found cognizant of control board and plant conditions. Control room and shift manning were in accordance with Technical Specification requirements. Posting for and control of radiation, contamination, and high radiation areas were appropriate. Use of and compliance with radiation work permits and use of required personnel monitoring devices were confirmed.

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 correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout, and lifted lead and jumper logs.

Pre-evolution briefings were noted to be thorough, with appropriate questions and answers. Operators displayed excellent knowledge of plant conditions. No unauthorized reading material was observed. Food, beverages, and hard hats were kept away from control panels.

2.2 Inoperable Drywell Hydrogen/Oxygen Analyzer

During normal operation, the drywell (primary containment) is inerted with nitrogen to preclude the presence of a combustible atmosphere. The primary purpose of the post-accident monitor (PAM) hydrogen and oxygen (H2/02) analyzers are to monitor post-accident combustible gas concentrations and in conjunction with the containment combustible gas control system (CCGCS), to control combustible gas concentrations in the drywell following a postulated accident. Technical specifications (TS) require both the capability to measure drywell hydrogen and oxygen concentrations and maintain their concentrations below specified limits under varied plant conditions. While continuous monitoring following a loss of coolant accident (LOCA) is required, only periodic verification of oxygen concentration and hydrogen analyzer operability are required during power operation. On a monthly basis during normal power operations, each PAM H2/O2 analyzer is verified to function correctly.

On September 12, 1992 the "A" H2/O2 analyzer failed surveillance procedure 8.M.3-13, "H2/O2 Analyzer System Monthly Functional Test." The licensee declared the "A" H2/O2 analyzer inoperable and entered a TS action statement which would require the reactor to be shut down if both H2 analyzers were not operable within seven days. The cause of the surveillance failure was identified to be air inleakage to the oxygen analyzer portion of the "A" H2/O2 analyzer. Maintenance personnel, employing several vendor recommendations, attempted to identify and correct the source of the air inleakage. However, they were not able to totally eliminate the air inleakage by the end of the fourth day. During this entire period, the licensee continued to record drywell oxygen concentration daily utilizing their normal sample point, the panel C41 continuous oxygen analyzer.

Licensing personnel determined that TS require that two drywell hydrogen analyzers must be operable. However, TS did not require that two drywell oxygen analyzers be operable. Oxygen concentration must be maintained <4% by volume during reactor power operation. In addition, oxygen concentration must be measured and verified below this limit at least twice per week. While the "A" PAM O2 analyzer was inoperable, the licensee maintained the capability to measure drywell O2 concentration with both the "B" PAM H2/O2 analyzer and the panel C41 continuous oxygen analyzer. On September 16, 1992, the "A" H2/O2 analyzer was returned to service and the hydrogen analyzer portion of procedure 8.M.3-13 was successfully completed. The licensee declared the "A" hydrogen analyzer operable and exited the TS action statement.

The inspector reviewed system descriptions and associated drawings, valve lineup configurations, and evaluated H2O2 monitoring component availability and license requirements for post-LOCA conditions. During inspector interviews with the licensee crew on watch, operators demonstrated a high level of knowledge regarding system response and post-LOCA oxygen and hydrogen monitor availability. Upon review of the requirements of 10 CFR 50.44(b), the Final Safety

Analysis Report, TS and associated safety evaluations, the inspector concluded that BECo had correctly verified operability of required equipment and properly exited the TS action statement.

3.0 RADIOLOGICAL CONTROLS (71707)

The inspector reviewed controls in place as well as the radiological condition of selected areas of the plant. Management tours of the radiological controlled area (RCA) continued to be thorough and directed toward minimizing personnel exposure. Survey postings, radiological conditions and controls were appropriate with no discrepancies noted by the inspectors. Additional training regarding use of robots at Pilgrim to reduce personnel exposure and upcoming changes to 10 CFR Part 20 "Standards for Protection Against Radiation" were noted to be excellent.

3.1 (Closed) Violation 90-25-01, Failure to Follow Radiological Controls Procedures

The initial violation cited three examples of the failure of onsite personnel to follow established radiological controls procedural requirements. Additionally, a fourth example of the same violation was identified in NRC Inspection Report 50-293/91-07. Each example had been identified by the licensee and each involved the failure of individuals to comply with properly established radiological boundaries and posting requirements. The events were of minimal safety significance but appeared related to occurrences identified by a previous NRC violation. Although the events were predominantly the result of personnel inattention to established controls, human factor considerations such as posting location, difficulty in observing posting changes, and redundant postings were identified as contributing factors.

In response to these events, senior licensee management increased the awareness of radiological protection requirements by including topics on these matters during weekly division manager meetings and monthly station safety meetings. A station survey was also conducted to identify and eliminate redundant postings and to clarify posting requirements. Further, on March 1, 1992, a major revision to the radiological work permit (RWP) program was implemented. The revision eliminated the use of "continuing RWPs" and instituted the use of a "dose tracker" mechanism. This change provided clear differentiation between low dose projected activities such as tours and inspections and more moderate dose projected activities that would require an RWP to accomplish. Inspector assessment of the RWP program revision is documented in NRC Inspection Report 50-293/92-03 (Section 3.1) issued on April 3, 1992.

Additionally, the licensee instituted the initial revision of procedure 1.3.106, "Conduct of Radiological Operations." Previously, radiological worker requirements had been well defined by general employee training, but onsite worker reference was somewhat complicated because standards and expectations were contained within several procedures. Inspector review of the conduct of this radiological operations procedure determined it provided a comprehensive, centralized instruction that delineated radiological controls requirements and the radiological protection program expectations at PNPS. The inspector concluded BECo's actions to address the cited violation were appropriate, and therefore this item was closed.

4.0 MAINTENANCE AND SURVEILLANCE (37828, 61726, 62703, 93702)

4.1 Maintenance Coordination during Main Condenser Backwash

On September 13 reactor power was reduced to support performance of a main condenser backwash to remove marine fouling and improve overall heat transfer performance of the main condenser. The licensee effectively utilized this period of reduced power operation to accomplish several pending maintenance activities. Planned maintenance included replacement of "B" recirculation pump motor generator (M-G) set brushes, repacking of the "A" condensate pump gland seal, plant heating system leak repairs, a control rod pattern exchange, and several surveillances.

Prior to the on-line recirculation M-G brush replacement, the training department built a mockup of the recirculation M-G and conducted several training sessions for electricians and electrical supervisors. The inspector attended the training sessions and noted that the mockup was an excellent training tool. In addition to very closely matching the actual M-G set configuration, special sensors were attached which would cause an alarm to annunciate and the mockup M-G set to trip if the brushes or pigtails were shorted during brush replacement. The alarm would identify the actual location where the short occurred. Training on the mockup resulted in several excellent comments and recommendations from technicians which improved the existing procedural instruction.

Following the completion of training, the licensee commenced the planned maintenance operation of replacing the "B" recirculation pump M-G set brushes on-line. The four "B" recirculation pump M-G set exciter end brushes were successfully replaced on September 13. Maintenance personnel were well briefed and completed the brush replacement promptly with no distractions or delays. On-line brush replacement eliminated entry into single recirculation loop operation and minimized control rod pattern manipulations. The inspector determined that the licensee approach to replacement of the "B" M-G set brushes on-line was technically sound. Additionally, the licensee demonstrated appropriate understanding of the importance of personnel training and worker feedback mechanisms in support of this maintenance.

In another area of planned maintenance, efforts to eliminate the excess leakoff from the area of the "A" condensate pump packing gland were unsuccessful. A closer inspection, after completion of gland repacking, determined that the packing gland was functioning properly. The source of the excess leakage appeared to be a deteriorated gasket at the stuffing box flange. Replacement of the gasket would involve major disassembly of the "A" condensate pump. The licensee decided that replacement of the gasket would be deferred to the upcoming maintenance outage, unless an alternate method to eliminate the excess leakage was developed sooner.

The "A" condensate pump was restarted and leakage was determined to be approximately 10-20 gallons per minute. This amount of leakage resulted in no operational degradation or limitations. The water was directed to the radioactive waste treatment system via the turbine building floor drains and equipment drains. The reactor achieved 100 percent power at 12:40 a.m. on

September 15. All maintenance scheduled for completion during the main condenser backwash, with the exception of the repair of the "A" condensate pump seal leakage, was successfully completed during the main condenser backwash. The inspector observed several of the maintenance activities including work on the "A" condensate pump seal assembly. Coordination between Operations Section and Maintenance Section personnel was effective in providing adequate work control.

4.2 Improved Electrical Connectors for Safety Related Components

High temperature detectors are installed on several safety-related systems to provide an initiation signal to isolate the systems in the event of a leak or break in the suction supply lines. Systems which have this protective feature include main steam (MS), high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), and reactor water cleanup (RWCU). Recent increase in the failure rate of the electrical terminal strip to which the temperature detectors are connected has resulted in several unplanned engineered safety feature (ESF) actuations or entry into Technical Specification limiting conditions of operation over the past few years. The licensee determined that the terminal block and connector failures were due to the repetitive torquing of the terminal block and connectors, associated with testing requirements. In addition, past use of inappropriately sized screwdrivers was noted to have resulted in damage to the terminal block in some instances.

The licensee developed a plant design change to eliminate the need to lift and land connector leads from the terminal block when replacing temperature switches during functional testing. A quick disconnect "grayboot" connector was developed for installation between the terminal block and the temperature switches. The installation was designed to provide a method to connect and disconnect temperature switches without lifting connector leads from the terminal block. This design change was first implemented on the HPCI system. Following installation, Instrumentation & Controls (I&C) personnel conducted a critique of the work to identify areas for improvement. The inspector observed the critique which identified several actions intended to reduce the system outage time needed to install the "grayboot" connectors on the remaining systems (RWCU, MS, and RCIC). Planned licensee actions included collection and verification of additional information during work package planning field walkdown, contingency work package planning for immediate repair of any damaged wiring or terminal blocks identified during "grayboot" installation, dedicated parallel support from electrical and quality control disciplines, and closer coordination with operations personnel to expedite post-work testing and system restoration. The critique was detailed, resulting in several useful recommendations which were effectively integrated into work packages for the remaining installations.

On September 21, 1992, the inspector observed installation (FRNs 92-02-11, 44, 45, 46) on the RWCU system. The prework briefing was excellent, including input from quality control and radiological protection representatives. When installing the first of eight temperature switches (TS-1291-14C), technicians noted that the terminal strip screw size found inplace differed from that specified in the work package. As a result of this finding, they questioned whether or not the installation torque value specified in the work package should be used. A nonintrusive visual walkdown had been performed for work package preparation since no drawing previously existed for this junction box. The difference between the two screw sizes was nondiscernible without actually removing screws from the terminal strip. The different screw size necessitated changing the lug size of the TS-1291-14C connectors. Replacement "Q" quality lugs were obtained and installed under the close supervision of a quality control representative. The I&C supervisor verified that based upon terminal block type, the screws found inplace were of the correct size and directed that the screws be torqued in accordance with procedure 3.M.3-51, "Electrical Termination Procedure", in lieu of the value specified in the work package. The terminal strip screws found inplace differed from that shown in the work package for three of the first four junction boxes. In each case, the discrepancy was correctly resolved in an expeditious manner. Technicians and quality control personnel displayed outstanding attention to detail and a questioning attitude throughout implementation of this design modification. Additionally, maintenance request log sheets were effectively used to provide detailed feedback for future maintenance activities.

Eight temperature switches with "grayboot" connectors were installed on the RWCU system during implementation of this design change. Self adhesive cable tie wraps were installed to reduce the strain placed on terminal connectors due to the weight of the cables. Four of the eight junction boxes had 3/4 inch elbow fittings which were widened to one inch as needed for installation of the new temperature switches. In addition, weep holes were drilled into the junction boxes to preclude moisture accumulation as discussed in IE Notice 89-63, "Possible Submergence of Electric Circuits Located Above Flood Level Because of Water Intrusion and Lack of Drainage." This design modification was well planned and was a positive initiative to improve the overall availability of several safety related systems. Recommendations resulting from the critique of the HPCI system installation were effectively implemented to reduce system outage time during subsequent installations (RWCU and RCIC). The licensee has scheduled the remaining "grayboot" installations to be performed during the midcycle maintenance outage scheduled to commence on October 24, 1992.

5.0 **EMERGENCY PREPAREDNESS** (40500)

On September 15, the licensee conducted a hazardous material drill. This was a joint participation exercise including the Plymouth Fire Department and Bourne Hazardous Material Team in addition to licensee personnel. On October 8, the licensee conducted a medical emergency drill involving transportation of a contaminated injured worker (simulated) to Morton Hospital located in Taunton, Massachusetts. Notification to the NRC was simulated by telephone call to the NRC resident inspector office. Both drills provided good training involving the joint response and interface between the BECo staff and several local support service agencies.

6.0 **SECURITY** (71707)

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 force staffing; vital and protected area barrier integrity; maintenance of isolation zones; behavioral observation; and implementation of access control including access authorization and badge issue, searches of personnel, packages and vehicles and escorting of visitors. Security force personnel continued to alertly perform their duties.

7.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (92701)

7.1 10 CFR 50.59 Safety Evaluation Process

The NRR Project Manager conducted a sample audit of the BECo 10 CFR 50.59 process. Safety evaluations and Plant Design Changes (PDC) associated with four evaluations were reviewed. The changes included 1) Dragon Excess Flow Check Valve Replacement, 2) Use of RHR Supplemental Fuel Pool Cooling, 3) IS1 Modification to Valve 10-CK-515 and 4) Addition of Scram Group/MSIV Logic to Control (Room) Panel C905.

The 10 CFR 50.59 process at Pilgrim is governed by controlled procedures that follow the guidance available in the Nuclear Management and Resources Council, Nuclear Safety Analysis Center, NSAC-125 "Guideline for 10 CFR 50.59 Safety Evaluations." Appropriate FSAR changes were made to describe the modifications implemented using the 50.59 process. The documentation associated with the process was comprehensive, as were the training modules used to instruct engineering and operations staff.

7.2 Potter & Brumfield Model MDR Rotary Relays

On January 6, 1992, the NRC issued Information Notice 92-04, "Potter & Brumfield Model MDR Rotary Relay Failures", that alerted licensees to the potential failure of this relay due to mechanical binding of the rotor due to coil varnish outgassing and chlorine corrosion. On March 2, 1992, the NRC also issued Information Notice 92-19, "Misapplication of Potter & Brumfield MDR Rotary Relays," that alerted licensees of a potential failure mode if the relays were used for switching direct current or low level loads. The notices were for information, and did not require licensee response or specific corrective action.

BECo reviewed the notices for applicability to Pilgrim Nuclear Power Station (PNPS), evaluating component and plant reliability databases, and concluded that Potter & Brumfield relays were not installed at PNPS. Based upon independent NRC review of BECo Operating Experience Review Program documentation, and discussions with cognizant system engineers, the inspectors similarly concluded that Potter & Brumfield MDR Rotary relays were not used at PNPS.

7.3 LER Review

7.3.1 RCIC Inoperability

Licensee event reports (LERs) 92-07 and 92-10, "Reactor Core Isolation Cooling System (RCIC) Inoperability," July 17, 1992 and September 17, 1992, respectively describe brief periods of RCIC inoperability. On June 18, 1992 and on August 18, 1992, during performance of quarterly

RCIC steamline area high temperature instrumentation functional testing, the licensee identified loose or stripped temperature switch terminal locations that required the system be declared inoperable to implement necessary repairs.

The steamline area temperature monitors would isolate RCIC in the event of a leak or break in the steam supply line. The temperature detection circuitry consists of two channels, each with two temperature switches in series. The quarterly surveillance sequentially disables each switch by installing jumpers, and verifies operation of the redundant switch by applying a heat source and observing energization of the trip or isolation relay. Over time, the repetitive landing and lifting of the test jumpers have caused the associated terminal block locations to weaken and become stripped. The identification of an increased number of stripped area temperature switch terminal block locations appears to indicate that age-related fatigue is the cause of the problem.

In response to this concern, the licensee initiated a design change to install "quick-connect" style electrical connectors that are suitable for repetitive manipulations. The design change is evaluated in Section 4.2 of this report. The LERs appropriately addressed the reporting criteria.

7.3.2 Target Rock Safety Relief Valve

LER 92-08, "Setpoint of Target Rock Safety Relief Valve Found Out of Tolerance," dated July 31, 1992, describes the discovery, as a result of testing conducted by Wyle Laboratories, that the as-found pressure setpoint of the pilot valve on a two-stage safety relief valve was not within the tolerance allowed by the PNPS Technical Specifications (TS). The LER also described the plant conditions at the time of and related to the removal of the subject safety relief valve for testing, and appropriately assessed the safety consequences, while discussing planned corrective actions.

The inspector reviewed several of the referenced documents listed in this LER, including PNPS procedures 3.M.4-6 and 8.5.6.2, problem report 92.0338, and an engineering evaluation (BECo letter no. 91-123) submitted to the NRC to comply with Technical Specification 3.6.D.4. The inspector confirmed that problem report 92.0338, with reference to a related problem report 91.0373, adequately addressed the need for enhancement of the procedure governing reassembly of the main steam relief valves and noted that proper handling of the safety relief pilot valves was included in the maintenance training program. The inspector also reviewed BECo letter No. 91-123 relative to the evaluation of safety relief valve leakage and the correlation of the resultant tailpipe temperatures to postulated setpoint drift. The inspector noted that this evaluation was accomplished utilizing General Electric reports on setpoint drift and tailpipe temperature correlation data, which represented the best available information at the time.

The setpoint problem identified in LER 92-08 represents a condition that was not evaluated in BECo letter No. 91-123. In particular, the steam leakage pathway caused by damage to the flexitallic gasket between the pilot assembly and the main valve body potentially created conditions of setpoint actuation beyond those allowed by the Technical Specifications, and yet within the tailpipe temperature correlations intended to ensure compliance with the Technical

Specifications. While the inspector did not question the licensee's planned corrective actions to these identified conditions, specifically as documented in problem report 91.0373, an issue was raised regarding consideration of the information gained from the test results related to LER 92-08 in future safety relief valve setpoint drift evaluations.

The PNPS TS 3.6.D.3 requires the licensee to perform an engineering evaluation during plant operation when two-stage Target Rock safety relief valve steam leaks raise the discharge tailpipe temperature above 212 degrees for more than 24 hours. The licensee agreed that steam leakage pathway identified in LER 92-08, as well as other possible leakage paths, should be considered in any future evaluations conducted pursuant to TS 3.6.D.3. The licensee is assessing the need for procedural guidance in this regard. Since LER 92-08 was appropriately handled from a regulatory standpoint and since the licensee has determined that the information gleaned from this event will be factored into future evaluations related to safety relief valve Technical Specification compliance, the inspector had no additional questions in this area. The setpoint drift of Target Rock two-stage safety relief valves has historically been an area of generic NRC interest (NRC Information Notice 88-30). Prior to restarting from the last refueling outage, all the Target Rock two stage safety relief valves were refurbished, retested and certified to be within specification. BECo program controls were found to be adequate to evaluate identified setpoint drift conditions and to ensure compliance with TS requirements.

7.3.3 **RCIC Isolation**

LER 92-09, "Automatic Closing of the Inboard Primary Containment System Group 6 Isolation Valve During Testing," dated August 14, 1992, describes the July 17, 1992, inadvertent automatic closure of the reactor water cleanup system (RWCU) inboard containment isolation valve during RWCU area temperature switch functional testing. The area switches temperature functions to isolate RWCU in the event of a leak or break in the suction line. The temperature circuitry is similar in design to that described in Section 7.3.1 regarding RCIC. In order to perform the temperature switch testing, the isolation function is bypassed by repositioning a keylock bypass switch from the NORMAL position to the HI TEMP. BYP position. The switch testing is accomplished by applying a heat source to the switch and by observing the de-energization of the primary containment isolation system (PCIS) relay. Following satisfactory conclusion of the test and re-energization of the PCIS relay, the keylock bypass switch is returned to the NORMAL position. During this event, approximately 10 seconds after returning the keylock bypass switch to the NORMAL position the inboard containment isolation valve automatically closed. The isolation was of minimal impact on plant operations and reactor safety. The operating RWCU pump tripped as designed in response to the isolation. The isolation was reset and the RWCU was returned to operation in less than one hour.

Subsequent BECo review identified no apparent cause for the isolation. The procedural instruction was determined to be accurate and to have been correctly implemented. The temperature switch was also found to be in proper calibration. The licensee concluded the most probable cause of the event to have been temperature switch instability due to residual test heat, sufficient to cause the switch to actuate. The surveillance procedure was revised to include a caution requiring that the temperature switch be allowed to cool for at least five minutes before resetting the keylock bypass switch to normal.

Independent inspector review of the event did not reveal any additional potential contributing causes to the isolation. The licensee investigation of the event was concluded to be effective.

8.0 ENGINEERING AND TECHNICAL SUPPORT (71707)

8.1 Scaffolding Controls

During inspection tours of several areas in the plant, the inspector observed the erection of scaffolds, planned for utilization during midcycle outage (MCO 9) work activities. The evaluation and control of scaffolding at PNPS is governed by Procedure No. 1.5.15, which delineates seismic requirements for scaffold erected in areas where safety-related components could adversely be affected. The inspector reviewed the current edition (revision 2) of this procedure and reviewed the Seismic Scaffold Control Log in the Control Room Annex to check the status of engineering evaluations for inprocess and completed scaffolds and to verify procedural adherence.

The inspector examined completed scaffolds and inprocess scaffolding activities in both the reactor building and plant auxiliary bays. Tagging controls, specified by procedure were checked, as were technical criteria (e.g., clearances) procedurally delineated to provide assurance that the completed scaffolds are seismically qualified. For one scaffold (i.e., control log no. 92-009) being erected in the "A" train, auxiliary bay, the inspector identified some specific clearance problems between the two-inch diameter scaffold tubing and both the positioner controls for temperature control valve, TV-3836, in the reactor building closed cooling water (RBCCW) pump piping header and the RBCCW suction header piping itself. The inspector discussed the clearance concerns with operations department supervisory personnel and noted that the discrepancies were immediately addressed and the scaffold was modified, where required. The inspector reinspected scaffold no. 92-009 after the "Seismic Scaffold" tag was attached and identified no nonconforming conditions.

Additionally, the inspector evaluated the process for scaffold review and approval, specifically assessing engineering involvement. Overall, an effective program to control scaffolding has been established and is being implemented at PNPS. However, it was noted that regular attention to the erection criteria prescribed in Procedure 1.5.15 is required to ensure total seismic qualification. At a routine plan-of-the-day meeting, the licensee discussed scaffold controls and distributed a controlled flow chart illustrating the process for scaffold review and approval. Given the level of scaffold erection ongoing in the plant in preparation for the upcoming mid-cycle outage, the inspector deemed the current level of management attention to scaffolding to be appropriate.

8.2 Operability Determinations

8.2.1 Training

On September 21, 1992, the inspector observed the conduct of training of BECo nuclear engineering division (NED) personnel in Braintree, Massachusetts relative to their programmatic approach to the resolution of degraded and nonconforming conditions. The training was presented by the Compliance Division Manager. This training, using the guidance provided in NRC Generic Letter (GL 91-18), provided an open forum for the discussion of the use of engineering judgment and design margin, as applied to operability determinations. A flowchart, representing the program established by BECo to evaluate degraded or nonconforming conditions at PNPS, was presented as part of the training. This flowchart illustrated the interrelationship between both operability and reportability determinations and the corrective action processes.

The inspector noted that the examples of component problems, presented in the training scenarios, were representative of past occurrences at PNPS. This teaching technique provided a realistic training vehicle. The inspector also observed good discussion relative to the roles of both the Technical Specifications and the FSAR in evaluating design margins and establishing the current plant licensing basis. The training was also provided to the onsite review committee, the Problem Assessment Committee, and the Operations and Technical sections. The inspector considered the conduct of this training to be a good initiative. Based upon the complexity and the interpretative nature of the concepts involved in the evaluation of GL 91-18, continued BECo management overview of this program is also warranted to appropriately monitor its effectiveness.

8.2.2 Valve Operability

On September 23, 1992, the licensee identified a wiring discrepancy in the control circuitry for motor operated valve, MO-1201-80. This valve is located in the discharge line of the reactor water cleanup (RWCU) system and is a primary containment isolation system (PCIS) component, isolating the RWCU system upon initiation of a Group VI signal. Since the RWCU system was already isolated for the conduct of preventive maintenance, no new active LCO's were entered. However, an additional valve in the discharge piping of the RWCU system was manually closed to compensate for the questionable status of valve MO-1201-80.

The licensee issued problem report 92.9176 and conducted an operability evaluation of the miswired valve condition. The inspector reviewed the technical section operability evaluation/recommendation, documented and approved on September 23, 1992, which recommended that valve MO-1201-80 be considered operable. The basis for this recommendation was the determination that the identified wiring configuration was electrically equivalent to the original valve wiring design. Since the valve also passed its previous surveillance testing, confirming the capability of automatic isolation in response to a PCIS Group VI signal, the safety function of the valve was verified to be operable.

The inspector noted that NED concurred with the station's electrical systems engineering staff in the operability recommendation. The inspector discussed the as-found wiring discrepancy, (i.e., an unauthorized jumper wire) with an electrical systems engineer. Electrical equivalency of the existing valve control circuit was determined by evaluating PCIS signal response across the jumpered contacts. PNPS drawings E217, sheet 65 (revision E7) and E226, sheet 107 (revision E6) were reviewed for the electrical connection details supporting the licensee analysis of electrical equivalency.

While the inspector had no questions regarding the operable status of valve MO-1201-80, the need for identification of the cause of the unauthorized jumper, as well as corrective action plans, were discussed. The licensee intends to perform the needed rewiring during plant conditions conducive to performing and testing the rework without risk of plant safety system challenges. Root cause analysis of the identified discrepancy will be addressed in the completed problem report. The inspector had no further questions in this area, and determined that the technical input for the operability evaluation of the nonconforming RWCU valve was appropriately detailed and complete. The RWCU system was restored to service on September 24, without incident.

9.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (30702)

9.1 Routine Meetings

At periodic intervals during this inspection, meetings were held with plant management to discuss licensee activities and areas of concern to the inspectors. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting on October 30th with licensee management summarizing the preliminary findings for this report period. No proprietary information was identified as being included in the report.

9.2 Management Meetings

On September 9, officials met with the licensee to receive a report on BECo's Self Assessment Program and to discuss activities planned for the upcoming maintenance outage currently scheduled to commence October 24, 1992. The meeting was conducted at the Chiltonville Training Center, in Plymouth, Massachusetts and was open to the public. The licensee's slide presentation, addressing the performance assessment and mid-cycle outage initiatives was previously enclosed as an attachment to NRC Inspection Report 50-293/92-16.