Enclosure

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

License No.	DPR-35
Report No.	96-10
Docket No.	50-293
Licensee:	Boston Edison Company 800 Boylston Street Boston, Massachusetts 02199
Facility:	Pilgrim Nuclear Power Station
Inspection Period:	November 23, 1996 - January 11, 1997
Inspectors:	R. Laura, Senior Resident Inspector B. Korona, Resident Inspector
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EXECUTIVE SUMMARY

Pilgrim Nuclear Power Station
NRC Inspection Report 50-293/96-10

This inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers resident inspection for the period of November 23, 1996 through January 11, 1997.

Operations: Extensive preparations, including simulator and classroom training, and management oversight led to a well controlled evolution to implement FFWTR. Reactor operators used excellent self verification and procedural adherence techniques. Some procedure changes were required for clarification shortly before the evolution indicating that the procedure preparer, reviewer, ORC members and training personnel missed earlier opportunities to identify and correct these issues. (Section O.4.1)

After significant changes were made to the problem report process, the training effectively explained the changes including a lower problem reporting threshold, a two level priority scheme, performing common cause analysis every 6 months and replacing the problem assessment committee with a corrective action review board. Meaningful exchanges of information occurred during questions from the training participants. (Section O.6.1)

Maintenance: Maintenance and I&C workers completed the HCU functional testing and Agastat calibration activities in a competent manner. Indications of overheating in normally-energized, Agastat relays were evidenced by the bobbin material and washers becoming brittle. A previous BECo evaluation, largely based on a Wyle lab test report, extended the service life from 10 to 22 years. These Agastat relay service life issues and generic preventive maintenance implications remain as **IFI 96-10-01.** Insulation resistance testing of the "B" core spray pump motor went very smoothly with no problems which was an improvement over past performance. (Section M.1.1)

Several lower level adverse equipment/material condition problems identified by the inspector were either went unnoticed or were incorrectly accepted by plant workers and members of the BECo management staff. For example, the CRD pump motor air inlet and outlet screens were partially clogged with dirt. Also, a steady 1/2 gpm packing leak on the condensate transfer jockey pump went undetected which contributed to radwaste in-leakage in the long term. The management tour implementation process yielded mixed results that was less than fully effective in ensuring the identification and correction of lower level equipment/material condition issues. (Section M.2.1)

A more rigorous approach of entering TS LCOs during surveillance tests, when required, better ensures compliance with TS requirements and also allows better consideration of risk management. Accordingly, **Unresolved item 50-293/95-26-01 is closed.** (Section M.3.1)

Effective reactor fuel receipt inspection training was provided to BECo personnel by a General Electric representative. The training was thorough and provided not only verbal direction and a videotaped presentation, but also "hands on" training on the refueling floor. Maintenance, reactor engineering, operations, and radiological protection personnel communicated well to perform the fuel inspections. Discrepancies were appropriately identified and dispositioned, which confirmed training effectiveness. (Section M.5.1)

LER 94-04 and its supplement were closed. The LER provided sufficient information pursuant to 10 CFR50.73 and NUREG 1022 involving a PCIS actuation during a RCIC surveillance test. (Section M.8.1)

Engineering: Engineering personnel completed an adequate safety evaluation generally bounding the effects of FFWTR of up to 75 degrees. The safety evaluation did not fully discuss two pertinent areas possibly affected by the change including the RIPDs and ATWS analyses. In one instance, engineers informally relied on verbal information from the vendor which was a poor practice. Subsequently, two vendor letters substantiated the 75 degree FFWTR operation assuring safe plant operations. The Operations Review Committee had previously approved the FFWTR safety evaluation and did not identify these weaknesses. **(UNR 96-10-02)** Two potential UFSAR update issues were noted regarding core operation in the MELLA region. **(UNR 96-10-03)**

Plant Support: No unusual or inconsistent increases occurred in the gaseous releases from PNPS to the environment during December 6 - 9, 1996. The main stack and reactor building ventilation 24 hour release totals were less than 1.0% of the TS limit and no unusual spikes occurred in the hourly average readings. Positive chemistry personnel performance was noted. (Section R.1.1)

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REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station (PNPS) began the period operating at approximately 100 percent rated power and remained at or near full power until January 11 when reactor power was lowered to approximately 65% to implement final feedwater temperature reduction (FFWTR). Operators commenced returning the unit to full power at the end of the period.

I. OPERATIONS

O1 Conduct of Operations[1]

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspector conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety conscious. During tours of the control room, the inspectors discussed any observed alarms with the operators and verified that they were aware of any lit alarms and the reasons for them. Any anomalies noted during tours were discussed with the nuclear watch engineer (NWE). For example, the 3-D monicore computer heat balance screen showed a feedwater subcooling parameter in an "abnormal" state as indicated by the red color of the display box. Operators could not readily explain the significance of this observation. Later, reactor engineers explained that although the heat balance calculation inputs into the thermal limit calculations, a bad data point would have been indicated by a magenta color box. The red box resulted from higher recirculation flow rates used near the end-of-cycle operation. The operations department manager included this information in the operations night notes. Other specific events and noteworthy observations are detailed in the following sections.

O4 Operator Knowledge and Performance

O4.1 Final Feedwater Temperature Reduction (FFWTR)

a. Inspection Scope (71707,93702)

A review was performed of the preparations, training and implementation of FFWTR conducted near the end of this operational cycle to increase cycle efficiency. A reduction of final feedwater temperature, obtained by securing extraction steam to the first and second point feedwater heaters, and opening the high pressure heater bypass valve (MO-3489), adds positive core reactivity to temporarily compensate for reactor fuel consumption. The inspector observed operator simulator training and reviewed a related written operator training lesson plan. NRC Region I engineers assisted the inspector in the technical review of the PNPS licensing, design bases information and the BECo 10 CFR

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[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

50.59 safety evaluation. During deep back shift inspection on January 11, 1997, the inspector observed operators implement the FFWTR method. The engineering aspects of the FFWTR are discussed further in Section III.E2.1 of this report.

b. Observations and Findings

The inspector reviewed the written operator training material prepared by operator training personnel for the operating crews. The training guide included a discussion of the pertinent portions of the PNPS licensing and design bases. Mentioned in the material was that BECo implemented FFWTR at PNPS in 1977, 1979 and 1983. However, the extraction steam was only removed from the first point heaters resulting in a feedwater temperature reduction of 30 degrees Fahrenheit. A larger feedwater temperature reduction of 60 degrees, not to exceed 75 degrees, was planned this cycle by removing the heating from both the first and second point heaters. The training covered the lessons learned from an event that occurred at another nuclear power plant in November 1995 where the operators failed to follow the procedure for implementing FFWTR. System engineers provided input how to best isolate the extraction steam to the first and second point feedwater heaters and how to open the high pressure heaters' bypass valve. Instructions were provided to remove the seal-in feature of related feedwater heater valves MO-3109, 3209 and 3489 to convert them into jog valves. This allowed for slow and controlled isolation of the first and second point heaters and opening of the bypass valve to control the addition of positive core reactivity in small increments.

In April 1996 operations support personnel modified (revision 8) Procedure 2.2.152, Feedwater Heater Extraction Steam and Heater Drains, and added Section 7.5 to provide instructions to accomplish the FFWTR. After simulator validation in October 1996, revision 9 was made to procedure 2.2.152. During this inspection period, the operating crew scheduled to conduct the FFWTR performed the evolution on the simulator. The inspector witnessed this final simulator dry run noting that the crew members conducted a thorough review of the procedural steps. A reactor engineer who had done this evolution in the past offered his insights to the crew. Two more procedural steps needed clarification. One concern identified by the operators was whether or not the feedwater temperature referenced was an average temperature of both trains or the lower temperature of either train. This was especially important when using a feedwater temperature limit graph in Attachment 4. A second concern identified by operators involved a note that stated not to take controls out of automatic when isolating the steam to the second point heaters. Operators felt that the note was too restrictive and should allow more flexibility when deciding to take manual control of the heater water level controls. After the training session, a procedure change was made to address the two aforementioned procedural issues. The inspector determined that the operating crew performed a thorough verification and validation of procedure 2.2.152.

The operating crew discussed the intent of an exception statement in the front of procedure 2.4.150, Loss of Feedwater Heating, which stated that the procedure was not applicable during the process of FFWTR. The inspector later expressed concern to the operations support team leader that the exception statement was confusing and did not transition well with procedure 2.2.152. A change was made to the exception statement which clarified when the procedure would be entered and used when implementing the

FFWTR method of 2.2.152. The inspector recognized the value of the questioning attitudes exhibited by the operators during the simulator training in the area of procedural adequacy; however, the operations support procedure preparer, reviewer, onsite review committee (ORC) members and operator training personnel missed opportunities to identify and correct these procedure quality issues at an earlier stage in the process. The inspector expressed concern to the plant manager that these procedure quality issues were consistent with the findings documented in NRC inspection report 96-80 and 96-08, and questioned whether a feedback loop exists to discuss the lessons learned from a procedure change quality point of view to improve future performance.

At the pre-evolutionary briefing (PEB) on January 11, 1997, the inspector observed substantial plant and executive level management presence in the control room. The operations department manager actively participated in the PEB with the operating crew. The nuclear watch engineer (NWE) and crew members pre-selected abort criteria for possible degradation in condenser vacuum and turbine vibration parameters. A reactivity manager was assigned to monitor the addition of reactivity as the feedwater temperature was lowered. The FFWTR method was implemented after reactor power was reduced to 66%. Extraction steam was isolated from the first and second point feedwater heaters in a very slow and methodical manner. The reactor operator closing the steam extraction valves continuously used the prescribed self verification process (i.e., STAR). The high pressure heater bypass valve was opened. A reduction of approximately 60 degrees in feedwater temperature was achieved with the FFWTR. After the heaters were isolated, the reactor was returned to 100% power with no problems. The inspector determined that operators conducted the FFWTR activities in a well controlled manner with excellent procedural adherence and self-checking techniques.

c. <u>Conclusions</u>

Extensive preparations, including simulator and classroom training, and management oversight led to a well-controlled evolution to implement FFWTR. Reactor operators used excellent self verification and procedural adherence techniques. Some procedure changes were required for clarification shortly before the evolution indicating that the procedure preparer, reviewer, ORC members and training personnel missed prior opportunities to identify and correct these issues.

06 Operations Organization and Administration

06.1 <u>Major Revisions to Problem Report Program</u>

a. <u>Inspection Scope (71707)</u>

The inspector reviewed training provided for a new problem report (PR) program contained in procedure no. 1.3.121.

b. Observations and Findings

During the training session, the new problem report process was discussed in detail using illustrative examples by the plant manager and operations support personnel who facilitated the training. The previous multilevel PR priority scheme was replaced by a bi-level significant condition adverse to quality (SCAQ) or condition adverse to quality (CAQ). A lower problem reporting threshold was established. Shortly after the training, the amount of problem reports initiated increased approximately three-fold with approximately 20-25 generated per weekday. Detailed root cause analyses will generally be performed for SCAQs. An apparent cause analysis will be done for CAQs that were caused by organizational or programmatic failures and/or human error/inappropriate actions. Every 6 months a common cause analysis will be performed to identify any adverse trends. Finally, the problem assessment committee (PAC) was eliminated and a new corrective action review board (CARB) was created. The CARB consists of a few select senior managers designed to review the adequacy of proposed corrective actions and their implementation for SCAQ conditions.

c. Conclusions

After significant changes were made to the problem report process, the provided training effectively explained the changes which included a lower problem reporting threshold, a two level priority scheme, performance of common cause analysis every 6 months and replacement of the problem assessment committee with a corrective action review board. Meaningful exchanges of information occurred following questions from the training participants.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726, 62707)

Using inspection procedures 61726 and 62707, the inspector observed portions of selected maintenance and surveillance activities to verify proper calibration of test instrumentation, use of approved procedures, performance of the work by qualified personnel, conformance to limiting conditions for operation, and correct system restoration following maintenance and/or testing. The following activities were observed:

- Station Blackout (SBO) EDG Operational Test

- Control rod drive hydraulic control unit (HCU) functional checks for level and pressure switches

- "B" core spray pump motor winding insulation resistance test

- Calibration of newly installed RPS Agastat time delay relays

b. Observations and Findings

Operators promptly secured the SBO EDG before the end of the two hour operational run due to the potential fire hazard caused by a leaking fuel oil pump injector tube. Operators closely followed the procedure when securing the engine. A small coolant leak due to a broken bolt in the coolant return header support identified by the inspector is further discussed in Section M2.1 of this report.

Instrumentation and controls (I&C) technicians competently calibrated new RPS Agastat relays and performed HCU level and pressure functional checks. Workers initiated PRs and promptly notified supervision of emergent problems. Activities were properly stopped at calibration transition points to facilitate shift turnovers. The normally-energized RPS Agastat relay vendor recommended service life of 10 years was extended by a WYLE lab test report no. 48687-REL-1.0 to 22 years which was subsequently accepted by BECo engineering in PR 94.0236. Inspections performed during this period revealed signs of overheating in the plastic relay bobbin material and end washers/spacers. The inspector observed the overheated material became brittle and crumbled easily. The extension of the 10 year qualification life to 22 years and potential generic implications for the preventive maintenance program constitutes an **inspector follow-item (IFI 96-10-01).**

Insulation resistance testing of the 4160 volt "B" core spray pump motor went smoothly with no problems which was an improvement from previous observations in an earlier inspection period. The operator wore a complete suit of protective clothing during breaker racking operations. Section II M1.1 of NRC inspection report 96-06 documented several difficulties encountered previously while testing the "B" RHR pump motor windings. At that time, the wrong megger box was brought to the work site and operators had to manually rack the breaker due to a broken test box and degraded connections.

c. Conclusions

Maintenance and I&C workers completed the HCU functional testing and Agastat calibration activities in a competent manner. Indications of overheating in normally-energized, Agastat relays were evidenced by the bobbin material and washers becoming brittle. A previous BECo evaluation, primarily based on a Wyle test report, extended the service life of these relays from 10 to 22 years. These Agastat relay service life issues and generic preventive maintenance implications constitute **IFI 96-10-01.** Insulation resistance testing of the "B" core spray pump motor went very smoothly with no problems which was an improvement over past performance.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 NRC Plant Tour Results and Evaluation

a. Inspection Scope (71707,62707)

Periodic field inspections were performed in selected plant areas, including the reactor building quadrant rooms, to assess the overall plant material condition. This review also included the results of plant tours made by the NRC Region I Administrator, Mr. Hubert Miller, on December 9, 1996 and NRC Chairman, Shirley Jackson, on December 19, 1996.

b. Observations and Findings

Progress was made during this period in the preservation coating of the control rod drive (CRD) quadrant room and in the outer auxiliary bay. However, the inspector identified equipment deficiencies in both areas that were not self-identified by plant workers or members of the BECo management staff. Three adverse conditions were noted in the CRD quadrant room. The CRD pump motor air inlet and outlet screens were partially clogged with dirt. Also, a yellow catch-containment installed under valve 301-40A was completely full of water due to valve packing leakage. The yellow catch-containment was readily visible from nearby locations. Lastly, the lower level CRD floor drain cover was missing which created the possibility of foreign material entering into the floor drain system. The inspector noted an NWE in the reactor building and showed the NWE the above adverse conditions for corrective actions. A few days later, the inspector verified that work request tags (WRT) were hung to address the CRD motor screens and the missing floor drain cover. The catch-containment under valve 301-40A had been drained of water and a tygon hose installed to allow the water leakage to drain to a nearby floor drain. Upon closer inspection, the inspector observed that water leakage from valve 301-40A was leaking on the floor; apparently when the catch containment was modified to install the drain hose, the catch containment was not sufficiently repositioned. The inspector reported this condition to the NWE and determined that increased attention-to-detail by radiological control technicians should have avoided this condition.

In the outer auxiliary bay area, the inspector observed packing leakage (approx. 1/2 gpm) from the condensate transfer jockey pump (P-111) was properly collected and directed to be processed as radwaste in-leakage. The morning station report tracked radwaste in-leakage as a performance indicator which averages approximately 30 gpm. The inspector noted that no WRT was hanging indicating that P-111 shaft packing leakage was not entered into the work control system for corrective action. Control room operators initially thought that the adverse condition was entered into the system. After a detailed equipment history review, the day watch engineer informed the inspector that no WRT existed for P-111 shaft leakage, but a PR and a WRT were initiated to obtain corrective actions based on the inspector's concern. Further review determined that a maintenance request (MR) was written on March 29, 1994 for the same problem but was later cancelled to engineering service request (ESR) 94-33. ESR 94-33 was written to replace the pump shaft packing with mechanical seals. The ESR was closed on November 29, 1994 with instructions to the maintenance staff to replace the packing with better material. A new MR was never written to perform the work. The inspector noted that the P-111 shaft leakage of approximately 1/2 gpm correlates to 21,600 gallons/month which contributed significantly to the radwaste in-leakage over a long period of time.

Some of the preservation coatings in the intake structure, performed as part of the plant material condition upgrade program, showed signs of degradation. Significant packing leakage (several gpm) was noted coming from the "A" salt service water (SSW) pump shaft which cascaded down to lower levels in the intake structure and ultimately into the water bays. Later in the inspection period, maintenance personnel adjusted the "A" SSW pump shaft packing to reduce the leakage. As mentioned in Section II M1.1 of this report, the (SBO) emergency diesel generator (EDG) was secured due to fuel oil leakage from high pressure tubing located between a fuel injector pump and injector. This leak was identified and closely monitored by the operators. Also during the SBO EDG run, the inspector identified a small coolant leak resulting from a broken bolt used to secure the cooling water return header support. The operators initiated a WRT to obtain corrective actions.

In the aggregate, these adverse conditions, combined with previously identified issues in 1996 NRC inspection reports, indicate that lower level equipment problems exist that were either accepted or unidentified by plant workers and management personnel. The inspector reviewed the 1996 management observation tour program results contained in electronic messages. Of the sample reviewed, the inspector noted that the quality of the department level manager results varied greatly, with very few performed in the last quarter of 1996. Evidence existed of executive level management participation as indicated by numerous tour reports. The management observation tour program expectations stem from various memoranda and electronic messages from the senior vice president nuclear dating back to September 6, 1994. Department level managers were expected to spend 4 hours in the plant each week and provide a write-up with an overall assessment.

Late in this inspection period, the licensee issued procedure IOTWI.002, Performance Evaluation Program, dated January 7, 1997. The new procedure provided guidance for management tours including review and trending of the tour results and the issuance of a quarterly trend report. The inspector noted that IOTWI.002 did not contain the expectations for the department level managers. The independent oversight team leader acknowledged the concern and informed the inspector that the management expectations would be issued soon.

c. Conclusions

Several adverse equipment/material condition problems identified by the inspector were either unidentified or incorrectly accepted by plant workers and members of the BECo management staff. For example, the CRD pump motor air inlet and outlet screens were partially clogged with dirt. Also, a steady 1/2 gpm packing leak on the condensate transfer jockey pump went undetected and contributed to radwaste in-leakage over the long term due to a work tracking oversight. The management tour implementation process yielded mixed results and was less than fully effective in identifying and correcting lower-level adverse equipment/material condition issues. A revised process was initiated.

M3 Maintenance Procedures and Documentation

M3.1 (Closed) Unresolved Item (50-293/95-26-01): Safety Equipment Operability During Surveillance Testing

a. Inspection Scope (61726)

A review was performed to assess BECo's evaluation and actions taken to address a surveillance test program weakness that allowed rendering safety equipment inoperable during surveillance testing and not entering the applicable limiting condition for operation (LCO).

b. Observations and Findings

Section 3.2.1 of NRC inspection report 50-293/95-26 documented that the evaluation of Request For Information (RFI) 90-175 appeared outdated, with improper interpretations of NRC regulations, based on the newer information contained in NRC Generic Letter 91-18. Subsequently, operations management reviewed NRC Generic Letter 91-18, Section 6.4, and related surveillance test information from other nuclear power plants. On November 15, 1996, the operations department manager issued operations section standing order 96-07, Revision O, which implemented a new and more rigorous approach for surveillance tests affecting LCOs. Operations support personnel prepared a detailed listing of LCO-related surveillance procedures that coded which ones require entry into a TS LCO. Based on standing order 96-07 and related program procedure changes, operators routinely enter LCOs during surveillance tests which render safety related equipment inoperable. The standing order was intended to remain in effect until revisions are made to the individual surveillance procedures. The inspector verified operators are using this standing order.

c. Conclusions

A more rigorous approach of entering TS LCOs during surveillance tests, when required, better ensures compliance with TS requirements and also allows better consideration of risk management. Accordingly, Unresolved item 50-293/95-26-01 is **closed**.

M5 Maintenance Staff Training and Qualification

M5.1 New Reactor Fuel Inspection Training and Performance (62707)

a. Inspection Scope

The inspector attended new reactor fuel inspection training provided to maintenance technicians and supervisors, reviewed applicable station procedures, and observed the fuel inspection activity. The inspector also observed interdepartmental communication during the fuel inspections to determine whether they were effective to complete the activity in a quality manner.

b. Observations and Findings

Fuel inspection training was performed by General Electric (GE) in early November. The training was attended by the maintenance technicians who performed the inspections, maintenance supervisors and radiological protection technicians. The training included a videotaped presentation of fuel fabrication and inspection as well as "hands on" training on the refueling floor. After the taped presentation, the GE representative further elaborated on the inspection techniques.

The "hands-on" training was also conducted by the GE representative. The inspector noted that technicians who had not performed the inspections before were grouped together for this training so that appropriate time and discussion was facilitated. The inspector verified that the training encompassed the inspections required by PNPS procedure 4.2, Inspection and Channeling of Nuclear Fuel. This training allowed not only training on the actual fuel inspection, but also an opportunity for the technicians to unload the fuel from the storage containers and move it with overhead cranes to the unloading station, inspection stand, and spent fuel pool.

The inspector observed portions of the inspections of the 208 fuel assemblies during this period and noted the inspections were performed in accordance with procedures 4.2; 4.1, Receiving and Handling of Unirradiated Fuel Assemblies; and 4.0, SNM Inventory and Transfer Control. Careful transportation of the fuel was observed including slow crane movement and positioning into the fuel inspection stand. Technicians were aware of the fuel movement and guided the bundles and channels into the stand, ensuring no unintentional contact was made that could potentially damage the fuel rods. Maintenance technicians were observed thoroughly inspecting the fuel in accordance with procedure. In addition, although it was only required by procedure to inspect the lower tie plate from the bottom of the bundles before they were placed in the inspection stand, this inspection was also performed after the bundles were channeled and before they were placed in the spent fuel pool. Maintenance technicians were observed carefully channeling the inspected fuel bundles before transportation to the spent fuel pool. Good coordination was observed between maintenance and the operator on the refuel bridge for the final transportation of the fuel assemblies to the spent fuel pool storage racks.

Maintenance personnel appropriately performed the dimensional inspections and identified a nonconformance in one of the bundles. Nonconformance report (NCR) 96-045 and PR 96.0577 were written to document fuel rod-to-fuel rod spacing gage sticking on one bundle. GE was contacted for guidance and the bundle was placed back in its shipping container until it was inspected again per GE instruction. The subsequent inspection was successfully performed and the fuel assembly was stored in the spent fuel pool.

RP technicians were observed performing the required surveys and smears during the fuel inspection activity. The maintenance supervisor on the refuel floor and reactor engineer on the refuel bridge communicated well and maintained proper special nuclear material inventory and transfer control in accordance with procedure 4.0. Material balance area (MBA) forms were appropriately completed to document fuel bundle movement from the storage crates into the spent fuel pool storage racks.

c. Conclusions

Effective reactor fuel receipt inspection training was provided to BECo personnel by a General Electric representative. The training was thorough and provided not only verbal direction and a videotaped presentation, but also "hands on" training on the refueling floor. Maintenance, reactor engineering, operations, and radiological protection personnel communicated well to perform the fuel inspections. Discrepancies were appropriately identified and dispositioned, which confirmed training effectiveness.

M8 Miscellaneous Maintenance Issues (92902, 92700)

M8.1 (Closed) Licensee Event Report (LER) 94-04: Automatic Closing of the Reactor Core Isolation Cooling System Turbine Steam Supply Isolation Valves During Surveillance Testing

The inspector reviewed LER 94-04, submitted to the NRC on September 2, 1994, and Supplement 1 to this LER, submitted on December 29, 1994, to verify accuracy, description of cause, previous similar occurrences, and effectiveness of corrective actions. The LERs were also reviewed with respect to the requirements of 10 CFR 50.73 and the guidance provided in NUREG 1022 and its supplements.

LER 94-04 reported an August 3, 1994 automatic primary containment isolation control system (PCIS) Group 5 actuation during the performance of a reactor core isolation cooling (RCIC) system quarterly surveillance test. Initial investigation of the PCIS actuation signal determined the direct cause was the failure of the governor control valve to respond to control system demand due to valve binding. Further investigation revealed the fulcrum dowel pins in the valve were misaligned, causing the valve binding. The supplement to LER 94-04 documented the results of BECo's root cause analysis of the improper alignment of the RCIC turbine steam governor control valve fulcrum alignment pins.

A detailed description of the event and BECo's troubleshooting activities was documented in NRC inspection report (IR) 50-293/94-18, Section 3.1. Unresolved item (UNR) 94-18-02 was opened pending BECo's evaluation of a discovered RCIC lube oil design issue and completion of the root cause determination for the governor valve failure. An NRC safety inspection of BECo's root cause analysis and corrective actions to address the entrapment of air in the RCIC lubricating oil system was documented in NRC IR 95-02.
UNR 94-18-02 was subsequently closed in NRC IR 95-15, Section 4.3.

The inspector verified that corrective actions that were planned when the unresolved item was closed were completed. Specifically, procedure 3.M.4-78, RCIC Turbine 5-Year Preventive Maintenance Inspection, was revised to include guidance on the alignment of the dowel pins during governor control valve reassembly. The drawing specified for revision in the LER and IR 95-15 was retired. The drawings which replaced it were verified to be revised to include the dowel pins.

LER 94-04 and its supplement accurately documented the event, troubleshooting activities, corrective actions taken and planned, root cause analysis results, the request for enforcement discretion while RCIC was inoperable longer than the allowed outage time in

BECo technical specifications, and similarity to previous events. The supplement was submitted in a timely manner after the root cause analysis was completed. The LERs properly addressed reporting criteria and corrective actions were completed. Therefore, **Licensee Event Report 94-013 and its supplement are closed.**

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 Safety Evaluation 3018: FFWTR

a. Inspection Scope (37551)

In Section I O4.1 of this report, the operational aspects of implementing FFWTR were discussed. The inspector reviewed the PNPS updated final safety analysis report (UFSAR) to identify any pertinent limits or related information that could be adversely affected by FFWTR. UFSAR Section 14.4, Abnormal Operational Transients, notes that FFWTR is a viable method to extend full power operation of the core by lowering feed-water inlet temperature and that the effects of the temperature decrease must be evaluated to determine any required adjustment to the operating limit minimum critical power ratio (OLMCPR) for the cycle. The PNPS core operating limits report (COLR), Revision 11D, contains MCPR operating limits with FFWTR in Table 3.3-3 for temperature reductions up to 75 degrees Fahrenheit. The basis for the COLR FFWTR information was a cycle-specific General Electric (GE) Analysis, WHB: 96-035 dated October 24, 1996, which also credited previous GE analyses for several cycle independent evaluations. Using the GE analyses, BECo completed 10 CFR 50.59 safety evaluation 3018, dated November 26, 1996, which concluded no unreviewed safety question (USQ) was involved with FFWTR of up to 75 degrees.

b. Observations and Findings

UFSAR Section 3.3.6.10, Impact of Increased Core Flow and FFWTR on Reactor Internal Components, states that reduced feed-water temperatures (analysis was done with a reduction of approximately 43 degrees Fahrenheit as compared to 75 degrees in SE 3018) increases the overall pressure differential across the reactor components in the high steam environments, such as the top guide, upper shroud, shroud head and steam dryer. The loads for these components are limiting at the reduced feed-water temperature condition. The inspector noted that Section "D", Affected FSAR Section, of safety evaluation 3018 listed UFSAR Section 14.4, but not UFSAR Section 3.3.6.10. The cycle-dependent GE evaluation stated that evaluation of the reactor internal pressure differences (RIPDs) was cycle independent; the assumptions and conclusions of which remained valid for Cycle 11. The inspector noted that SE 3018 appeared inconsistent with UFSAR Section 3.3.6.10 because SE 3018 allowed FFWTR up to 75 degrees whereas UFSAR Section 3.3.6.10 assumes approximately 43 degree reduction.

Engineering personnel later informed the inspector that they were aware of the wording contained in UFSAR Section 3.3.6.10 but chose not to document the basis for accepting the 75 degree FFWTR on LOCA RIPDs in writing because General Electric personnel stated over the telephone that the increase on the RIPDs was very small relative to the margin that existed in the design and that the change was intended for only this operating cycle. The inspector considered reliance on verbal information from a vendor in lieu of providing a written basis in SE 3018 for LOCA RIPDs was informal and a poor practice. Further, the statement in SE 3018 that the changes incurred by FFWTR did not affect any cycle-independent assumptions did not appear to be correct. Engineering personnel held further discussions with GE who provided a letter dated January 16, 1997 substantiating the prior verbal information provided to BECo.

Also, the inspector noted that UFSAR table Q7-1 in the SE 2842 package, Initial Conditions For ATWS Analyses, listed feedwater temperature as 367 degrees (normal value). The inspector questioned engineering personnel how FFWTR of up to 75 degrees affected the ATWS analyses since SE 3018 did not discuss this aspect. Engineering personnel recontacted General Electric who issued another letter, dated January 21, 1997, which indicated that the long-term phenomena of ATWS was dependent on initial core power and not initial feedwater temperature. These two engineering issues (i.e., RIPDs and ATWS) associated with SE 3018 constitute **Unresolved item 96-10-02** pending further NRC review.

Further, the inspector questioned licensing personnel about two UFSAR update issues. UFSAR section 14.4.2, Operating Flexibility Options referenced the core operating flexibility option no. 1 which involved the extended load line limit (ELLL) rather than the current maximum ELLL (MELLL). The regulatory affairs department manager informed the inspector that the UFSAR update for MELLL should address MELLL in UFSAR Section 14.4.2. The inspector obtained a copy of SE 2842, approved by ORC on December 1, 1994, which did not update the aforementioned section. Also, the inspector questioned why the previous UFSAR update submitted in 1996 pursuant to 10 CFR 50.71(e) did not include SE 2842. Licensing personnel informed the inspector that UFSAR changes were not updated until the modification was made and then all related drawings changes were made as part of modification close-out. On January 15, 1997, Beco completed a regulatory relations group self assessment (96-4) that identified several opportunities for improvement with updating the UFSAR. For example, the current process of waiting until modification close-out before updating the UFSAR does not provide timely support to the organization. The inspector expressed concern that the existing UFSAR update process has the potential to result in violations of 10 CFR 50.71(e). These UFSAR update issues constitute **unresolved item 96-10-03** pending further NRC review.

c. <u>Conclusions</u>

Engineering personnel completed an adequate safety evaluation generally bounding the effects of FFWTR of up to 75 degrees. The safety evaluation did not fully discuss two pertinent areas possibly affected by the change including the RIPDs and ATWS analyses. In one instance, engineers informally relied on verbal information from the vendor which was a poor practice. Subsequently, two vendor letters substantiated the 75 degree FFWTR operation assuring safe plant operations. The Operations Review Committee had previously approved the FFWTR safety evaluation and did not identify these weaknesses. **(UNR 96-10-02)** Two potential UFSAR update issues were noted regarding core operation in the MELLA region. **(UNR 96-10-03)**

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Gaseous Activity Release Review

a. Inspection Scope (71750)

A review was performed of main stack, offgas and reactor building ventilation gaseous release data from December 6 - 9, 1996 to verify compliance with technical specification (TS) limits. During this time period, the reactor operated at or near full power.

b. Observations and Findings

The inspector obtained an EPIC plant computer printout for the hourly averages of gaseous activity for the reactor building ventilation release rate (microcuries/sec.), offgas release rate (MR/HR) and main stack release rate (microcuries/sec.). Chemistry personnel routinely add the hourly average readings each day to develop a total 24 hour release number which was compared to TS limits. Inspector review of the hourly average readings detected no unusual or inconsistent increases in the gaseous release data for 12/6-9/96. In all instances, the 24 hour total release data for main stack and reactor building ventilation was much less than 1.0% of the TS release limits.

c. Conclusions

No unusual or inconsistent increases occurred in the gaseous releases from PNPS to the environment during December 6 - 9, 1996. The main stack and reactor building ventilation 24 hour release totals were less than 1.0% of the TS limit and no unusual spikes occurred in the hourly average readings. Positive chemistry performance was noted.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on February 6, 1997. The licensee acknowledged the findings presented.

X3 Management Site Visit Summary

On December 9 and 10, Mr. Hubert Miller, NRC Region I Regional Administrator visited the site to meet with the resident inspectors, interview several members of the licensee staff and tour the plant. Mr. Richard Conte, Region I DRP Branch 5 Chief, visited the site on December 11 to provide routine oversight activities of the resident inspectors and to interface with BECo managers. On December 19 NRC Chairman Shirley Jackson visited the site for a plant tour and meeting with senior BECo managers. Mr. Hubert Miller accompanied the NRC Chairman during her visit on December 19.

X4 Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for additional verification that licensees were complying with Updated Final Safety Analysis Report (UFSAR) commitments. For an indeterminate time period, all reactor inspections will provide additional attention to UFSAR commitments and their incorporation into plant practices and procedures. While performing inspections discussed in this report, inspectors reviewed the applicable portions of the UFSAR. Several UFSAR related issues were identified and documented in Section III E2.1 of this report.

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 82301:	Evaluation of Exercises for Power Reactors
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND UPDATED

Opened

IFI 96-10-01	Agastat Relay Service Life Extension and Generic Implications
UNR 96-10-02	SE 3018 (FFWTR) Weaknesses
UNR 96-10-03	UFSAR Update Concerns For SE 2842 (MELLA)

Closed

UNR 95-26-01	Entering TS LCOs During Surveillance Tests
LER 94-04	Automatic Closing of the Reactor Core Isolation Cooling System Turbine Steam Supply Isolation Valves Due to High Steam Flow Signal During Surveillance Testing

LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
APRMs	Average Power Range Monitors
BECo	Boston Edison Company
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CS	Core Spray
EP	Emergency Preparedness
EPIC	Emergency and Plant Information Computer
ESF	Engineered Safety Feature
gpm	gallons per minute
I&C	Instrumentation and Controls
IFI	Inspection Follow-Up Item
IR	Inspection Report
LER	Licensee Event Report
MG	Motor Generator
MR	Maintenance Request
NCV	Non-Cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NWE	Nuclear Watch Engineer
PNPS	Pilgrim Nuclear Power Station
PR	Problem Report
RHR	Residual Heat Removal
RP	Radiological Protection
SALP	Systematic Assessment of Licensee Performance
SE	Safety Evaluation
SNM	Special Nuclear Material
SRO	Senior Reactor Operator
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
WWM	Work Week Manager