U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 96-02

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

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: February 28 - April 6, 1996

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EXECUTIVE SUMMARY

Pilgrim Inspection Report 96-02

Safety Assessment/Quality Verification:

- Licensee Event Report 96-01, dated April 4, 1996, which reported a licensee discovery of an inoperable differential pressure switch designed to actuate on a low pressure condition in the torus relative to the reactor building, was closed this period. (Section 6.1)

- There was some evidence of procedure quality and usage issues being identified, corrected and trained upon during this period. Opportunity remains to minimize the potential for future consequential problems such as the scoop tube lock-up event that occurred during this period. (Section 6.2)

Plant Operations:

- An inadvertent lock-up of the "A" recirculation pump MG set scoop tube occurred due to an inadequate surveillance procedure and a missed opportunity by the licensed operator to prevent this event. However, the immediate operator response was effective in preventing a more serious event such as a reactor scram. This event clearly demonstrated the importance of identifying and improving procedure quality since this surveillance activity had been completed successfully numerous times before. (Section 2.2)

- Operators continued to identify potentially safety significant equipment problems during plant tours and initiated proper corrective actions. This was evidenced by the identification of the air leak from valve AO-7011B located in the Torus Room and also HCU 18-23 which became degraded. Two minor issues involving problem identification and resolution were identified involving non-safety related air regulator leaks; a mounting problem not readily obvious went undetected in the first instance, and past experience was not promptly applied to determine the problem scope by operations personnel in the second instance. (Section 2.3)

- A review of the tagging program found the program requirements well understood and implemented by operations personnel. A relatively low number of active and safety-related tagouts existed, indicating that tagouts were generally closed when related work was completed. A review of audit reports and problem reports indicated only a few minor and isolated performance problems. An operations management initiative exists to streamline the tagging process improve work efficiency. (Section 2.4)

Maintenance and Surveillance:

- Plant worker response to a failed reactor building-to-torus vacuum breaker was effective in evaluating the cause and repairing the failed differential pressure micro-switch. An innovative temporary modification was thoroughly written and reviewed to ensure plant and personnel safety. Adherence to approved procedures and the work plan were effective overall and appropriate post-work testing was completed to return the vacuum breaker to service and exit the LCO in a timely manner. (Section 3.2)

- Operators promptly declared the RCIC system inoperable after a failed operability surveillance on April 3 when test data entered in the required action range. Maintenance troubleshooting activities were well planned and controlled. Excellent cooperation and communication was maintained between the system engineer, I&C engineers and supervisors, work control personnel, and operations. Personnel systematically tested the flow controller, EGM, and electric governor relay (EGR) portions of the RCIC Woodward governor speed control system. Prior to returning the system to service, a thorough discussion of all observed anomalies and possible questions was completed. (Section 3.3)

- Extensive outage preparations occurred during this period to facilitate the upcoming mid-April shutdown maintenance outage. One minor inconsistency in the UFSAR was identified concerning the recirculation pump mechanical seal design life. (Section 3.4)

- Two minor examples of improper electrical termination torquing were observed this period resulting from improper procedural usage and confusing procedural guidance on the use of torque tables. A quality control inspector identified this concern, and the inspector observed the problem during a subsequent activity. Planned corrective actions appear appropriate. (Section 3.3)

- The program controls were adequate to ensure that preventative maintenance (PM) tasks were properly tracked and completed as scheduled or deferred when necessary with proper technical evaluation and justification. The tracking and trending of feedback from the PM program was limited. (Section 3.5)

- The program controls implemented for on-line maintenance, which included the recent addition of weekly limiting condition for operation (LCO) maintenance review committee meetings, were judged to be good with several initiatives noticed such as weekly LCO board meetings. The program for conducting on-line LCO maintenance is fairly new at Pilgrim; the effectiveness of which has not been fully evaluated. The three on-line LCO maintenance activities observed were well conducted reflecting knowledgeable personnel, good coordination and well staged tools and parts. One unresolved item was opened (**UNR 50-293/96-02-01**) for BECo management to evaluate the adequacy of their procedure guidelines for performing safety-related maintenance activities and to ensure management expectations in this area are clearly delineated. This is a program weakness. (Section 3.6)

Engineering:

- A review of the spent fuel pool licensing basis, performed by the NRR Project Manager, determined that PNPS has no implicit or explicit prohibitions that prevent a full core offload during refueling outages. The current PNPS licensing basis is the BECo submittal of record that supported License Amendment 155. An updated final safety analysis report (UFSAR) update will be submitted by BECo to include a maximum fuel pool temperature of 142 degrees F and 6.41 hours of time to boil which are limiting parameters for the current refueling offload analyses. This will remain as an inspector follow item (**IFI 50-293/96-02-02**) (Section 4.1)

- Region-based specialists completed the Engineering portion of the core inspection program, concluded that BECo had programs and administrative controls in place that provide the basis for a sound corrective action program. A previous NRC violation involving SBM switches was closed-out. Progress was noted in the reduction of the historical backlog of action items related to problem reports and operating experience and vendor technical manual reviews. An engineering self assessment activity for field revision notices (FRNs) was considered a positive initiative. Quality assurance conducted thorough audits in the engineering area. (Section 4.2)

Plant Support:

- Chemistry personnel responded well to the identification of the higher indications of silica in the spent fuel pool. Personnel communicated with other licensees to obtain industry information and place BECo's data in perspective. When site visits and historical data review has been completed, further recommendations will be considered. (Section 5.1)

- Overall radiological safety performance was very good. The radiological problem report process continues to provide excellent identification of radiological incidents. In some cases, critical evaluation of root cause and subsequent assignment of corrective actions was lacking. The calibration of radiation protection instrumentation was well controlled. Some calibration procedural enhancements were identified as an opportunity for improvement. Two violations of regulatory requirements were identified. One violation (**VIO 50-293/96-02-03**) concerned an unlocked calibration source. The second violation (**VIO 50-293/96-02-04**) concerned the release of radioactive material to an unlicensed battery recycling facility. External exposures were tracking below the goal of 97 person-rem for the year and no internal exposures were recorded. Pilgrim's personnel exposures were high for 1995 as compared to the industry boiling water reactor average. Radiological safety problems identified by the licensee were declining. (Section 5.2)

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1.0 SUMMARY OF FACILITY ACTIVITIES

Pilgrim Nuclear Power Station (PNPS) operated at approximately 100% of rated power throughout this inspection period.

On April 3, 1996, operators made a formal notification (Event Number 30257) to the NRC headquarters operations officer to report that the reactor core isolation cooling (RCIC) system was declared inoperable after a review of quarterly surveillance test data identified discharge pressure in the "action required" range. Also computer traces identified oscillations on the flow controller when it was in automatic control. The report was made pursuant to 10CFR50.72(b)(2)(iii)(D). On April 5, BECo declared the RCIC system operable. Further details are discussed in Section 3.3 of this report.

2.0 PLANT OPERATIONS (71707, 92901, 93702)

2.1 Plant Operations Review

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and focused on safety; specific events and noteworthy observations are detailed in the following sections.

2.2 Recirculation Pump Motor-Generator Set Scoop Tube Lockup

On February 28, with the plant operating at 100% power, an inadvertent lockup of the "A" recirculation pump motor-generator (MG) occurred during the performance of procedure 8.C.7, Weekly (BOP) Equipment Check, Revision 17, dated December 16, 1994. The inspectors reviewed the operations actions taken in response to this event and interviewed the operator involved and the operations support personnel tasked with evaluating the event.

During the local performance of procedure 8.C.7, Attachment 1, Section J, Equipment Check For the Recirculation MG Set DC Oil Pumps, alarms were received in the control room indicating a scoop tube lockup occurred. Operators promptly entered procedure 2.4.19, Recirculation Pump MG Set Scoop Tube Lockup. The Nuclear Watch Engineer directed all involved personnel to report to the control room and verified that the actions taken during the performance of procedure 8.C.7 caused the observed control room alarms. A specific brief was then held to reset the lockup per procedure 2.4.19 and re-perform the DC oil pump equipment check per procedure 8.C.7. These actions were completed satisfactorily with no further problems experienced and Problem Report (PR) 96.9063 was initiated.

Each recirculation pump has two AC oil pumps, one is normally running and one in standby, and one DC pump. The function of the DC pump is to start upon a trip of the running AC pump to help the recirculation pump MG set coastdown after a trip. Procedure 8.C.7, Attachment 1, Section J directed the operator to procedure 2.2.84, Reactor Recirculation System, to perform equipment checks on the Recirculation MG set DC oil pumps. The operator reviewed procedure 2.2.84 and copied the only section which referenced the DC oil pumps, Section

7.1.1 [16] Plant Prestart Checks. The purpose of procedure 8.C.7 was to verify the DC pump would start and run when manually started, known as "bumping" the DC pump. However, procedure 2.2.84 directed the operator to secure the running AC pump and verify the DC pump automatically started. As a result of this action, oil pressure dropped causing the scoop tube to lockup.

A critique was conducted in accordance with procedure 1.3.63, Conduct of Critiques and Investigations, to gather facts, ensure the plant was in a safe condition, and implement immediate corrective actions. The inspector reviewed the critique documentation and verified appropriate personnel attended the critique. At that time, the plant was in a safe condition and the immediate corrective actions had been taken.

Operations support evaluation determined the direct cause of the event was an inadequate procedure with a contributing cause of human performance/infrequent task. The inspector verified that the procedure was changed to provide clearer guidance on what was required for this test and the inappropriate reference to procedure 2.2.84 was deleted. In addition, operations management issued a Night Order reinforcing the concept of STAR (stop-think-act-review) to ensure that the actions being performed will not cause undesirable effects on the plant or operations. Also, the night order cautioned that procedures which haven't been performed by an individual for some time require extra attention.

The inspector reviewed procedures 8.C.7 and 2.2.84 and confirmed that there was not a proper section in 2.2.84 to perform a DC oil pump bump and the operator did go to the only section of that referenced procedure that discussed the DC oil pumps. The inspector discussed this event with the operator who indicated that although this is a weekly test, he had not performed this test in approximately one year. The operator also indicated that he strictly followed the procedural guidance. The inspector discussed the event with other operators and the evaluator to determine why no other operator had this same problem. Apparently, other operators were more familiar with procedure 8.C.7 understanding that only a bump was needed and procedure 2.2.84 was not required. All agreed that the procedure if strictly followed was inadequate. The inspector notes that since this procedure had been completed successfully numerous times before, those operators missed an opportunity to identify the procedural inadequacy and then correct the procedure which would have prevented this event. The procedure revisions were appropriate to prevent recurrence.

The inspector concluded that the "A" recirculation pump MG set scoop tube lockup resulted from an inadequate procedure combined with operator failure to self check and verify his actions were appropriate. The immediate operator actions taken were effective in preventing a more serious event. The event critique and PR processes were appropriately used and the implemented and recommended corrective actions appear to be appropriate to prevent recurrence. However, at the time of the event, procedure 8.C.7 was inadequate constituting a violation of 10 CFR 50, Appendix B, Criterion V, Procedures, which requires that activities affecting quality shall be prescribed by procedures appropriate for the circumstances. This licensee-identified and corrected violation is dispositioned as a Non-cited Violation in accordance with Section

VII of the NRC Enforcement Policy. Additionally, NRC concerns pertaining to procedural usage are further discussed in Section 6.2 of this report.

2.3 Problem Identification and Resolution

(a) Inspection Scope.

The inspector assessed the degree that operators identified plant equipment deficiencies and initiated corrective action. Equipment deficiencies noted by the inspector during plant tours were checked to see whether or not the deficiency had been properly identified and entered into the work control system. The initial screening of problem reports conducted at the plant manager's morning meeting was also monitored.

(b) Observations and Findings.

A low problem reporting threshold existed as evidenced by the initiation of several problem reports (PR) and/or maintenance requests each day. One potentially significant problem identified by an operator was air leakage from the actuator of the drywell equipment sump primary containment isolation valve, AO-7011B, which is located in the Torus Room. After promptly initiating PR 96.9104, the immediate operability review determined that AO-7011B remained operable since the valve was designed fail closed upon a loss of air. Hence, the containment isolation function of the valve remained available. A second potentially significant problem identified by operators was the degraded performance of hydraulic control unit (HCU) 18-23 which required frequent recharging. Operators declared HCU 18-23 inoperable, followed the applicable technical specification requirements and promptly initiated PR 96.9142. The inspector witnessed mechanical maintenance technicians replace HCU 18-23 during this assessment period, and noted the workers were effective in correcting the deficiency. The inspector determined that operators routinely identified equipment-related issues and initiated corrective action.

During a tour of the refueling floor, the inspector heard air leaking from an air pressure regulator which supplies air to Panel C-67, for ventilation damper controls. Upon closer inspection, the inspector observed a work request tag (WRT) hanging on the regulator which was initiated by the systems engineer. The inspector observed that two regulators were installed in parallel such that one could be isolated with manual isolation valves. The inspector questioned operations management why the leaking air pressure regulator was not isolated as a good operational practice. Operations management was aware of previous operational experience with these regulators and referenced a problem analysis data sheet (PADS) dated March 10, 1993. The PADS (MSED 93-118) provided a method to determine whether or not the regulator failed or if the two regulators in parallel have slightly different setpoints resulting in air bleeding down. Subsequent to the inspector questioning the issue of allowing the air leakage to be uncorrected, the method used by the PADS was employed and the regulator determined to be failed. The work-it-now (WIN) team subsequently replaced the regulator.

On another plant tour, an NRC Region I manager identified that mounting fixtures for another air pressure regulator for panel C-61A in the reactor building were pulled from the wall. Although a WRT hung on the regulator identified an air leak, the loose mounting of the air line tubes was not identified. The inspector determined that the air lines were not safety related or seismically qualified. Subsequently, operations personnel initiated a WRT to obtain corrective maintenance.

(c) **Conclusions.**

Operators continued to identify potentially safety significant equipment problems during plant tours and initiated proper corrective actions. This was evidenced by the identification of the air leak for AO-7011B located in the Torus Room and also HCU 18-23 which became degraded. Two minor issues involving problem identification and resolution were identified involving non-safety related air regulator leaks. A mounting problem not readily obvious went undetected in the first instance, and past experience was not applied to determine the problem scope by operations personnel in the second instance.

2.4 Tagging Program Review

The inspector reviewed BECo's tagging program to determine whether tagouts were implemented in accordance with the station tagging procedure, scope of active tagouts, and to verify operations effectiveness during tagout reviews.

The inspector reviewed procedure 1.4.5, PNPS Tagging Procedure, Revision 41, dated August 14, 1995. A total of 74 active tagouts were listed with less than 30 percent affecting safety-related equipment. Several older tagouts were examined and determined to be properly evaluated and still required. A total of 13 tagouts were identified as originating before 1994 with the oldest dated November 1989. The inspector field-verified several tagouts including those for the station blackout diesel, control rod drive system, and drywell coolers. Also, a check was done to verify the need for the tagouts still existed. No discrepancies were identified. The inspector verified that caution tags and danger tags were used appropriately, tags were legible and placed on correct components, and the intended electrical and mechanical isolations were complete. Monthly and the 1995 annual tagging audits, done per procedure 1.2.4, Operations Section Performance Assessment Program, were complete and in accordance with procedural requirements. Operations management informed the inspector that a major rewrite of the tagging program is underway to improve work efficiency.

The review of the tagging program found the program requirements well understood and implemented by operations personnel. A relatively low number of active and safety-related tagouts existed, indicating that tagouts were generally closed when related work was completed. A review of audit reports and problem reports indicated only a few minor and isolated performance problems. An operations management initiative exists to streamline the tagging process.

3.0 MAINTENANCE AND SURVEILLANCE (61726, 62703, 92902)

3.1 Routine Maintenance and Surveillance Observations

The inspector observed portions of the following selected maintenance and surveillance activities:

- 8.7.2.1 Measurement of Standby Gas Treatment Filters and Fan Capacity

- MR 19503326 Station Blackout Diesel Generator Frequency Meter Calibration and Replacement

The work was performed by knowledgeable personnel in accordance with applicable procedures and work packages. Instrumentation was properly calibrated and used where applicable. The inspector verified conformance to limiting conditions for operation and appropriate post-work testing. Systems were returned to their normal configuration after testing and maintenance.

3.2 Inoperable Reactor Building-to-Torus Vacuum Breaker

During the performance of quarterly surveillance procedure 8.M.3.4, Reactor Building (RB) to Suppression Chamber Vacuum Breaker Sensor Calibration, instrumentation and controls (I&C) technicians observed the failure of a differential pressure switch to automatically trip. This condition would have prevented the automatic actuation of reactor building-to-torus vacuum breaker, Train B, under a low pressure condition in the torus relative to the reactor building.

PNPS has two 100 percent capacity reactor building-to-torus vacuum relief trains. Each train (i.e. vacuum breaker) consists of an air operated relief valve (AO-5040A/B) and an in-series passive check valve (X-212A/B). One function of these valves is to equalize the pressure between the torus and reactor building to maintain structural integrity of the containment. In addition, the AO-5040 valves also serve a containment isolation function. This function was not affected by the discovered failure in the "B" vacuum breaker.

Maintenance troubleshooting and engineering judgement determined the cause of the switch failure was the fusing of the output contacts of a micro-switch located within the differential pressure switch. The inspector reviewed temporary modification (TM) 96-04: Maintenance Repairs on PISD-5040B, and attended a related operations review committee (ORC) meeting convened to review the TM. The modification prevented the "B" vacuum breaker from opening during the micro-switch replacement by installing two jumpers across the instrument output contact. The modification was well prepared and included a thorough safety evaluation and design adequacy review. The ORC discussions were open and thoroughly addressed plant and personnel safety.

The inspector attended the pre-job brief and observed I&C technicians replace the failed micro-switch. The briefing was detailed and included a discussion of the applicable limiting conditions for operation (LCO), temporary modification, work to be performed, foreign material exclusion controls, and possible problems that may have been encountered. The jumper installation and location were discussed with the technicians and control room operators and personnel were staged in case of a problem. The appropriate maintenance request package and procedures were at the job site and in use. Management oversight and, although not required by the procedure, quality control (QC) presence was noted. The new switch was properly installed and satisfactorily post-work tested. Good overall conduct and control of the evolution was observed.

Although procedural usage was effective, QC review of the completed work package identified improper torque used on the terminal screws. PR 96.9118 was issued to document and evaluate this discrepancy. Although the initial QC evaluation documented that the screws were under-torqued, further discussion with engineering determined that the screws were overtorqued by approximately 2 inch-pounds. This condition discrepancy was reviewed in an engineering evaluation that determined the installation was adequate and did not result in inoperability of the system. The corrective actions identified in the PR included the revision to procedure 3.M.3-51, Electrical Termination Procedure, to include torquing specifications for the identified type of terminal block installed and training of I&C and E-lab technicians and supervisors on the proper use of the torque tables in the procedure. The inspector discussed these corrective actions with the PR evaluator and questioned whether any further clarification of the procedure would be made. The evaluator said that clarifications to the procedure will be considered. The inspector determined the corrective actions were reasonable.

Plant personnel response to a failed reactor building-to-torus vacuum breaker was effective in evaluating the cause and repairing the failed differential pressure micro-switch. The temporary modification was thoroughly written and reviewed to ensure plant and personnel safety. Although electrical terminations were improperly torqued following the installation, adherence to approved procedures and the work plan were effective overall and appropriate post-work testing was completed to return the vacuum breaker to service and exit the LCO in a timely manner.

3.3 Unplanned Reactor Core Isolation Cooling (RCIC) Inoperability

The inspector observed (locally in the RCIC turbine room) the quarterly performance of procedure 8.5.5.1, Attachment 1, RCIC Pump Operability Flow Rate and Valve Test at Approximately 1000 psig, on April 3, 1996. During this test, a control room operator determined that the RCIC turbine controller was not controlling properly and took manual control. As mentioned in Section 1.0, after the test was completed and inservice test (IST) data and system computer traces were reviewed, the licensee declared the system inoperable and entered the associated LCO. PR 96.9156 was issued to evaluate the root cause for RCIC discharge pressure which was in the "required action" range.

The inspector responded to the control room with the system engineers after the control room operator took the controller to manual. The inspector observed proper evolution control while the system was monitored. At this time, although no abnormalities were observed in the control room when the system was returned to automatic, the decision was made to perform Attachment 2 to the procedure to try to repeat the abnormal conditions seen by the operator. Attachment 2 is essentially identical to the Attachment 1 test originally performed; however, Attachment 2 does not require IST data to be obtained. The inspector reviewed the computer traces with the system engineer and determined that although the initial swings of the controller observed by the operator were expected, the traces indicated controller output oscillations during the second run per Attachment 2. These oscillations were of concern to the system engineer.

The inspector observed portions of maintenance troubleshooting activities. The work was done in accordance with maintenance request (MR) 19600786 and approved troubleshooting, system operating, and calibration procedures. Excellent cooperation and communication was observed between the system engineer, I&C engineers and supervisors, work control personnel, and operations. Personnel systematically tested the flow controller, EGM, and electric governor relay (EGR) portions of the RCIC Woodward governor speed control system. These troubleshooting activities determined that the initial controller instability was due to the EGR portion of the speed control circuit starting to fail. The EGR's function is to convert an electrical signal, from the flow controller through the EGM control box, into mechanical movement of the servo/governor valve through the porting of hydraulic oil. The licensee replaced the EGR and sent the removed EGR to the vendor for further evaluation and determination of a root cause for the EGR failure. Prior to returning the system to service, a thorough discussion of all observed anomalies and possible questions was completed.

During the troubleshooting performed within the EGM cabinet, a degraded lug was identified on a terminal of the EGM. The licensee amended the troubleshooting procedure to include re-lugging this wire. The inspector observed I&C technicians perform this activity. The wire was properly re-lugged, reinstalled, and checked for continuity. After re-termination of the wire, the inspector discussed the torquing requirements with the technicians and I&C supervisor who also observed the activity. In light of the recent QC finding on torque values discussed in Section 3.2, the inspector questioned the use of one section of the torque table in procedure 3.M.1-34, Generic Troubleshooting and Maintenance Procedure, versus another. The supervisor agreed that the requirements were confusing and engineering was consulted. The final determination was that the wire needed only to be terminated hand-tight per a calibration procedure for that component and the torque value used was sufficient. Although no operability concern existed, this additional example further substantiated the QC concern over proper use of the torque tables contained in 3.M.1-34 and 3.M.3-51 (section 3.2, above). The same tables are contained in both procedures. The inspector discussed these findings with I&C management and determined that the training planned as corrective action for the problem report generated on the vacuum breaker and the licensee's agreement to consider procedure clarifications appear to envelope this problem.

On April 3, operators made a formal notification (Event Number 30257) to the NRC headquarters operations officer to report the RCIC system inoperability. The report was made pursuant to 10CFR50.72(b)(2)(iii)(D). Following appropriate post work testing, the RCIC system was declared operable on April 5 remaining in the 14 day LCO for 29 hours. The licensee plans to submit LER

96-03 for this run failure in accordance with the requirements of 10CFR50.73.

Operators properly declared the RCIC system inoperable after a failed operability surveillance on April 3 due to IST data falling in the "required action" range for differential pressure. The subsequent troubleshooting activities were well planned and controlled. Excellent cooperation and communication was maintained between the system engineer, I&C engineers and supervisors, work control personnel, and operations. Personnel systematically tested the flow controller, EGM, and electric governor relay (EGR) portions of the RCIC Woodward governor speed control system. Prior to returning the system to service, a thorough discussion of all observed anomalies and possible questions was completed. The operations department manager requested engineering to provide a summary document of the troubleshooting efforts and root cause to assist the nuclear watch engineer in returning the system to an operable status.

3.4 Preparations for Mid-April Outage

The inspector monitored the planning, preparations and training during this inspection period to support the planned maintenance outage scheduled for April 19 - 23, 1996. The planned shutdown was scheduled to replace a leaking safety relief valve (i.e., RV 203-3B) and replace the mechanical seal cartridge on the "A" recirculation pump due to seal leakage. Other corrective maintenance, inspections and modifications were also planned.

Four BECo personnel, representing operations, maintenance, radiological protection and work management, went to another boiling water reactor (BWR) to observe the conduct of a relatively short duration refueling outage. The trip was intended to identify improvements that could be made at PNPS. After observing the refueling outage, the multi-disciplined team returned to PNPS at the beginning of this inspection period. Several meetings were held to discuss the work control practices observed and a report was issued on March 8, 1996. The team identified twenty-five recommendations that could be adopted at PNPS in the work control/outage management area. For example, the concept of using work planning teams for significant work activities was viewed in a positive manner. Also, the outage schedule at the peer BWR facility better integrated man loading requirements, support group activities and the outage windows were developed by work planning teams. The inspector concluded that dispatching a 4 member team of BECo workers to monitor the performance of a peer BWR plant was an excellent self assessment activity. Tentatively, BECo management plans to implement several new concepts, such as the work planning teams to develop outage equipment windows and staffing of a centralized work planning and management location, during the upcoming planned outage.

The inspector attended maintenance training sessions held for the replacement of RV-203B and the "A" recirculation pump mechanical seal replacement activity. The mechanical seal training took 3 full days to complete. System engineering, maintenance crews and radiological control personnel attended the training. Mock-ups of the mechanical seal and recirculation pump housing (borrowed from Millstone) were used during the training. Also, BECo purchased the latest version of the vendor manual to ensure the incorporation of the latest vendor recommendations. Maintenance supervisors provided oversight and expectations for the maintenance workers. Additionally, BECo contracted the pump vendor to provide expertise in the seal rebuilding and replacement activities. An actual seal was used under vendor instruction which allowed the work crew to disassemble, rebuild and replace an actual seal. The inspector noted that this allowed maintenance workers the opportunity to refine the necessary skills to measure and establish the prescribed close tolerances within the mechanical seal. During the classroom portion of the training, the instructor reviewed briefly PR 95.9246 concerning the use of an incorrect revision of the maintenance procedure when rebuilding the current installed mechanical seal cartridge for the "A" recirculation pump. An outstanding action item in PR 95.9246 was to determine whether or not using the incorrect revision contributed to the leaking mechanical seal.

The crew rebuilt the spare seal approximately seven times with the last two times under full mock-up conditions including dress-out in anti-contamination clothing. After completion of the training, the maintenance crew rebuilt the seal to be used during the upcoming outage. The vendor participated with the crew during the actual rebuilding activity. A leak check of the rebuilt seal was performed using a test rig with satisfactory results. The training and preparation for rebuilding and replacing the "A" recirculation pump mechanical seal was considered a strength.

The inspector reviewed Section 4.3, Recirculation System, of the updated final safety analysis report (UFSAR). The description of the recirculation pump seal design was generally accurate. One minor UFSAR description that needed to be updated was the description of the expected seal life. The UFSAR states "Each seal is designed for a life of one year based on a 90% probability factor." The newer design used during the last several years has an expected life of approximately four years (i.e., two operational cycles). The inspector informed the system engineer and engineering management of the incorrect UFSAR wording. BECo indicated to the inspector that the wording would be updated to reflect the longer seal life.

Three hours of classroom training was provided for replacement of the pilot assembly for RV-203B. No safety relief valve mock-up was available to be used for training. The maintenance training instructor used diagrams and reviewed the actual work plan. Good training for replacement of the pilot valve assembly was observed.

The inspector concluded that extensive outage preparations occurred during this period to facilitate the upcoming mid-April shutdown maintenance outage. Especially noteworthy was the 3 full days of maintenance training, including mock-ups, to rebuild and replace the leaking "A" recirculation pump mechanical seal cartridge. Vendor participation in the training and during the actual seal rebuilding activities, along with the use of the latest vendor technical manual, contributed to high quality maintenance training. The high quality training resulted directly from worker participation and management involvement. Positive interaction between maintenance workers and the radiological control staff was observed. One minor inconsistency in the UFSAR was identified concerning the recirculation pump mechanical seal design life. The use of enhanced work management practices such as the use of work planning teams in developing the outage schedule windows, observed by BECo personnel recently at another boiling water reactor and implemented at Pilgrim for the upcoming outage, is an example of a high quality self assessment.

3.5 Preventive Maintenance Program

(a) Inspection Scope

The preventive maintenance (PM) program controls identified in Procedure 1.8, "Master Surveillance Tracking Program" were reviewed for adequacy. In addition, interviews with staff program administrators were conducted, and PM deferral records as well as trending and tracking of PM feedback were also reviewed.

(b) Observation and findings

The electrical and mechanical maintenance groups were issued weekly reports that listed all PMs with an 8-day look ahead sorted by due and drop dead date. In addition, special PM report notices were issued weekly for PMs in the alert status (i.e., past due date), priority status (i.e., last-minute notice for those PMs about to go past the drop dead date) and failure to comply status (i.e., past drop dead date).

Operations and instrumentation and calibration (I&C) were issued daily reports that provided an 8-day look ahead for technical specification surveillances as well as the special report notices described above for the maintenance groups.

There were 73 deferred PMs tracked as of March 14, 1996. The inspector noted the running deferral rate ran approximately 2.5% of the total non-technical specification population of 3,417 PMs. A review of a sample of approximately 20 of these deferred PMs indicated most were not being deferred for long periods. However, the inspector found one example of an emergency diesel generator PM (P001273) deferral that was deferred for 1 month without providing a technical bases for deferral. The PM was a basic inspection with 3 major tasks that included lubricate fuel racks, take a glycol sample and operate the dc emergency fuel pump. The most recent Procedure 1.8 Revision issued October 23, 1995, added Attachment 5 for PM deferrals, that provided the engineers a standard format for evaluation and documentation. This was a program enhancement that should provide improved documentation if properly utilized.

No performance indicators have been published for PMs since October 1995. Procedure 1.8, Section 5.5.2[3] stated that deferrals will be tracked as a performance indicator. The maintenance manager indicated he was planning to implement performance indicators to trend PM performance in the near future.

Three technical specification surveillance tests were missed in 1995; none have been reported in 1996 to date.

Tracking and trending of feedback from the PM program was limited at this time. System engineers tracked significant PMs deferred or overdue surveillance as part of their system status criteria checklists, but did not trend and track PM feedback. The nuclear plant reliability data system (NPRDS) looked at generic failures within the site and trended against industry performance for specific systems. Several examples of improvements implemented this past year as a result of this program were reviewed (e.g., the affected components included dc station batteries, hydraulic control units, and drywell cooler valves).

There were 250-280 PM items that were moved from RFO-11 and were being planned for performance prior to the outage and another 60 items that were being considered for deferral to RFO-12. The BECo staff was still evaluating the scheduling/planning/execution of these items at the time of the inspection.

(c) **Conclusions**

PM tasks were scheduled with appropriate priority for completion. Program controls identified in Procedure 1.8, "Master Surveillance Tracking Program" were adequate to ensure that PM tasks were properly tracked, scheduled (future view) and completed as scheduled or deferred when necessary with proper technical evaluation and justification. PM deferrals were relatively low indicating good control of the program. Tracking and trending of feedback from the PM program was limited. No concerns were identified in this area.

3.6 On-Line Maintenance Program

3.6.1 Program Controls

(a) **Inspection Scope**

The on-line LCO maintenance program controls identified in Procedure 1.2.2, "Administrative Ops Requirements" were reviewed for adequacy. The weekly LCO maintenance review committee meeting and several daily work week manager/scheduling meetings were attended.

(b) **Observations and Findings**

Procedure 1.2.2 was recently revised on March 8, 1996, to add further controls to the LCO planning check list. The additional controls provided were judged to be good. Training still needed to be conducted on the revised procedure. The work control manager had planned to complete the training by May 1, 1996.

The LCO maintenance review committee meeting scheduled to be held every Thursday reviews the on-line LCO maintenance planned for the following week. This was a new initiative and only the second time this meeting was held. The meeting appeared to be an excellent initiative to ensure all material, plans, and post-work testing were adequate to support the planned work. The meeting was thorough and several changes in required post-work testing were initiated during the meeting. The Pilgrim staff were still resolving some of the details of the process as evidenced by the questions that were asked during the meeting.

The conduct of daily work week manager/scheduling meetings had improved significantly from those observed during the last outage (see NRC inspection report 50-293/95-09). The meetings were conducted in a professional manner and the daily schedule was followed with few exceptions. In interviews with several maintenance supervisors and managers from both operations and maintenance departments, the comments indicated that the new work control process was well received and viewed as a significant improvement over the old processes.

(c) **Conclusions**

The program controls for conducting on-line maintenance with the addition of the new weekly LCO maintenance review committee were judged to be good with several initiatives noted such as the weekly LCO review board meetings. The program for conducting on-line LCO maintenance is fairly new at Pilgrim, so the effectiveness could not be evaluated.

3.6.2 Performance of On-Line Maintenance

(a) **Inspection Scope**

The inspector observed some portions of the following three on-line LCO maintenance activities and reviewed the completed work packages at completion of the work.

- Maintenance request (MR) # 1950-3494, disassembly of MO-1001-37B, inspection of motor actuator for binding of tripper finger with casing and modification if required.

- MR# P9500215 overhaul and rebuild of Bettis actuator for valve AO-5036B Torus purge supply isolation valve.

- MR# P9500216 overhaul and rebuild of Bettis actuator for valve AO-5036A Torus purge supply isolation valve.

(b) **Observations and Findings**

The work observed was well supervised by the cognizant supervisors. The workers appeared to be knowledgeable when questioned. Tools and parts were well staged. Good coordination between engineering and operations was also observed. No problems were noted with the actual execution of the work.

(c) **Conclusions**

The inspector concluded the work was well conducted reflecting knowledgeable personnel, good coordination with engineering and operations, and tools/parts well staged.

3.6.3 Work Control Processes for Safety-Related Work

(a) Inspection Scope

The program requirements for the preparation of maintenance work instructions/procedures used to perform safety-related maintenance were reviewed to ensure management expectations were properly and consistently implemented. The review focused on the program controls for minor and major maintenance, the use of vendor manuals, and when an operations review committee (ORC) approved procedure was required.

(b) Observations and Findings

The inspector reviewed portions of two on-line safety-related preventative maintenance activities (i.e., MR# P9500215 and P9500216, 5-year overhaul and rebuild of Bettis actuators for valves AO-5036B and AO-5036A torus purge supply isolation valves). The work was accomplished using a vendor manual in lieu of an approved procedure. In observing the work in progress, the inspector noted that the mechanics did not use a documented method to ensure all required steps of the vendor manual were performed. The inspector further noted a number of steps in the technical manual that, if improperly performed or omitted, would potentially affect component operation/reliability. The inspector concluded that the mechanics who performed the work were knowledgeable and performed the work without any detected problems. However, the inspector was nonetheless concerned that critical performance steps, that could be considered beyond the skill of the craft, could be overlooked in the future without sufficient work process controls and concluded this was an area of vulnerability.

The inspector reviewed BECo Procedure 1.5.17, "Conduct of Maintenance," Revision 3, Section 6.10, "Adherence to Procedures" which stated completion of all steps of a maintenance/I&C surveillance procedure shall be signified by one of the following methods as required: information entry; check mark; initials; signatures; data entry; N\A or N\P. The maintenance requests (MRs) referenced the use of vendor manuals V1014 and V0354 for performance of the work, but contained only limited guidance, signifying methods, and verifications for actual work performance within the MRs (e.g., torque adapter plate bolts to 140-150 ft-lbs and QC to verify torque wrench).

The inspector also reviewed BECo Procedure 1.5.20, "Work Control Process," Revision 3, Section 7.4.2 which specified that work plan details shall be commensurate with the complexity of the task. The inspector noted that the procedure provided clear and detailed controls for the performance of minor maintenance. However, the procedure did not provide clear guidance for preparing work plan instructions to conduct normal maintenance activities. There was no guidance provided for the level of detail required for the preparation of maintenance work instructions and when an operations review committee (ORC) approved procedure was required based on the safety significance and/or the complexity of the planned work. For example, the emergency diesel generator monthly preventive maintenance procedure, 3.M.3-61.1 was an ORC-approved procedure, but the safety-related PMs on the Bettis actuators described above were not ORC-approved procedures.

The plant manager, maintenance manager and work control manager acknowledged the inspector's concern, indicating that a review would be done to evaluate the adequacy of BECo's procedure guidelines for performing safety-related maintenance activities and to ensure management expectations in this area were clearly delineated. BECo planned to issue a revision to Procedure 1.5.20 by May 1, 1996. In the interim, the BECo work control manager planned to review each job on a case-by-case basis to ensure critical steps were clearly specified in the procedure or MR. BECo indicated that the proposed procedure change will specify that when planned safety-related corrective or preventive maintenance is performed using a vendor manual for guidance, verification steps will be inserted into the work instruction to ensure that critical elements of the maintenance are verified to have been performed. BECo further indicated that an evaluation of the guidance provided for the level of detail required for the preparation of normal/major maintenance work instructions would be performed along with when an ORC-approved procedure was required based on the safety significance and/or the complexity of the planned work.

(c) **Conclusions**

The inspector concluded that the program requirements for the preparation of maintenance work instructions/procedures were not clearly defined to ensure management expectations were properly and consistently implemented. This program weakness is **(UNR 50-293/96-02-01)** pending NRC review of the BECo re-evaluation.

3.6.4 Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR descriptions.

While performing the inspections discussed in this report section, the inspector reviewed the applicable portions of the UFSAR that related the areas inspected. The inspector verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters. For example, the wording in BECo Procedure 1.5.20, "Work Control Process," Revision 3, Section 7.4.2 which specified that work plan details shall be commensurate with the complexity of the task agreed with Pilgrim's updated safety analysis report, Section 13.6, "Station Procedures."

4.0 ENGINEERING (37551, 92903)

4.1 Spent Fuel Pool Licensing Basis Review

4.1.1 System Design

The NRR Project Manager reviewed the spent fuel pool licensing basis including technical specifications, updated safety analysis report (UFSAR) and all related information sent to the NRC to support license amendments. This special review resulted from recent experience in this area at another Region I reactor facility. The spent fuel storage pool is a stainless steel lined, reinforced concrete structure that has been designed to withstand earthquake loading as a Class I structure. Technical Specification Amendment 155, issued June 22, 1994, authorized the storage of 3859 spent fuel assemblies in the spent fuel pool. Interconnected drainage monitoring channels are provided behind the liner to permit leakage monitoring and free gravity drainage to the floor drainage sump. The passage between the spent fuel storage pool and the refueling cavity above the reactor vessel is provided with two double sealed gates with a monitored drain between the gates.

Unintentional draining of the pool during refueling with the gates open is prevented by design with no penetrations below approximately 10 feet above the top of the stored spent fuel and with lines extending below this level equipped with siphon breakers to prevent siphon backflow. There are two skimmer surge tanks to accommodate placement of large items in the pool such as a spent fuel cask. Makeup water is normally transferred from the condensate storage tanks to the skimmer surge tanks to make up for normal fuel pool losses. The maximum makeup rate using this method is 200 gpm. Additional water could be provided through the fuel pool spargers from system interconnections (condensate transfer pumps or residual heat removal [RHR] system pumps) to the fuel pool cooling system. Water could also be provided from fire hose stations supplied by the fire water pumps. (reference UFSAR Section 10.3)

The Fuel Pool Cooling and Cleanup System (FPCC) is designed to maintain the fuel pool temperature at or below 125°F during normal and refueling operations and maintains fuel pool water clarity through filtration and demineralization. FPCC is comprised of two pumps, two heat exchangers, one filter, a filter backwash system, a demineralizer, indications and alarms, and associated piping and valves.

In order to dissipate the heat load of a normal refueling off-load (28% of the core), the heat exchangers are sized for a combined heat load of 6.3E6 Btu/hr @125°F. Core off-loads in excess of 28% up to full core off-load can be accommodated by interconnection of the RHR system to the FPCC system. The RHR/FPCC intertie line was sized to remove a heat load of 23E6 Btu/hr at 1200 gpm. There are two modes of operation for the interconnection of RHR and FPCC. Mode 1 is a combination of RHR and FPCC and Mode 2 is RHR only. These modes are explained in more detail in the licensee's UFSAR Change Request, associated with plant design change (PDC) 94-37, for UFSAR Sections 4.8 and 10.4.

Use of RHR for augmented fuel pool cooling is also addressed in UFSAR Section 4.8.5.6.

4.1.2 Current Licensing Basis Summary

(a) Technical Specification Requirements:

3.10.C Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the pool water level shall be maintained at or above 33 feet.

4.10.C Spent Fuel Pool Water Level

Whenever irradiated fuel is stored in the spent fuel pool, the water level shall be recorded daily.

5.0 Major Design Features:

5.5 Fuel Storage:

B. The Keff of the spent fuel storage pool shall be less than or equal to 0.95.

C. Each fuel assembly in the spent fuel pool shall have a maximum K-infinity less than or equal to 1.32 and an enrichment of 4.6% U-235 or less averaged over the axial planar zone of highest average enrichment.

D. The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 3859.

E. Loads in excess of 2000 lbs. shall be prohibited from travel over fuel assemblies in the spent fuel storage pool.

F. No fuel which has decayed for less that 200 days shall be stored in racks within an arc described by the height of the cask around the periphery of the energy absorbing pad.

(b) Maximum heat load in the pool under refueling outage conditions is limited to design analysis input value of 27.0E6 Btu/hr (HOLTEC International Report HI-92925, "Pilgrim Nuclear Power Station Spent Fuel Storage Capacity Expansion"' dated December 1992.) In the safety evaluation for Amendment 155, dated June 22, 1994, the NRC verified the decay heat values determined by the licensee and found the values to be conservative.

(c) The fuel pool temperature is limited 142°F (primary cooling mode) for all planned refueling outages including full core offloads. This analyzed temperature was at end-of-life with the core offload and operating in augmented fuel pool cooling (Mode 2).

Under the worst case emergency condition for loss of all means of forced fuel pool cooling, time to boil was analyzed to be 6.41 hours. This provides enough time for plant operations to introduce alternative cooling methods or to supply water to replace such loss to assure maintenance of the spent fuel pool level. (Amendment 155, Safety Evaluation, dated June 22, 1994)

(d) The use of RHR to augment spent fuel pool cooling is discussed in the UFSAR and the HOLTEC International Report reviewed by the staff as part of the licensee submittal for the Amendment 155 and is considered part of the CLB. The licensee has used RHR for augmented fuel pool cooling since RFO 6 (1983-1984). The licensee's Operating Procedures applicable to normal and augmented operation of cooling for the spent fuel pool were reviewed.

(e) A delay time before fuel transfer of 120 hours was used in all fuel transfer analyses. There were other assumptions made any of which could alter the time to transfer. The licensee intends to use the maximum bulk temperature of 142°F and 6.41 hours time to boil as limiting parameters in fuel offload analyses for the primary method of cooling during future refueling outages. An update to the UFSAR will reflect these limits. Additionally, the licensee will perform a safety analysis for the RF011 core offload to ensure they remain within their licensing bases.

(f) There are no implicit or explicit prohibitions within the CLB against performing a full core offload for any given refueling outage.

4.1.3 Conclusion

The review of the spent fuel pool licensing basis, performed by the NRR Project Manager, determined that PNPS has no implicit or explicit prohibitions that prevent a full core offload during refueling outages. The current PNPS licensing basis is the BECo submittal of record that supported License Amendment 155. An updated final safety analysis report (UFSAR) update will be submitted by BECo to include a maximum fuel pool temperature of 142 degrees F and 6.41 hours of time to boil which are limiting parameters for the current refueling offload analyses. Some areas being evaluated by BECo for update follow.

(a) TS 3.10.C and 4.10.C require that the spent fuel pool level be maintained at or above 33 ft and that the level shall be recorded daily whenever irradiated fuel is stored in the spent fuel pool. A sampling of operator logs for the two previous refueling outages and normal operating logs were reviewed for compliance. Levels were properly recorded and within TS limits for all logs reviewed.

The TS Bases page regarding spent fuel pool water level, 3.10.C, was not clear for the reference frames for pool level and reasoning for establishment of the minimum level.

TS section 5.5.D limits the number of fuel assemblies in the spent fuel pool to 3859. This limit is sufficient for the current license of the plant. The licensee does not have all the racks in the pool to allow for the number of authorized assemblies. Additionally, there is an agreement between the licensee and the Commonwealth of Massachusetts that requires further analysis of alternate storage means before all the racks can be placed in the pool.

The TS Bases pages for Section 5.5, Fuel Storage, do not provide insight for most of the Major Design Features TS. This was identified to the licensee for consideration in future bases page revisions.

(b) The licensee prepared a Spent Fuel Pool Cooling and Demineralizer System Mechanical Design Basis, Report Number MDBR11-E1. The design heat load (MBtu/hr) for two of the conditions exceed those calculated in the HOLTEC report. In the condition with similar operating equipment (both SFPC systems in operation) the fuel pool temperature does not exceed 142°F and in the condition with a single SFPC system in operation the temperature of the fuel pool reaches 195°F which is less than boiling. The design heat loads differences are accounted for by the differences in assumptions. BECo clearly understood that the CLB for Pilgrim is the submittal of record considered by the NRC staff in issuing Amendment 155.

(c) The licensee will submit an UFSAR update that will include a maximum fuel pool temperature of 142°F and 6.41 hours time to boil as limiting parameters for refueling offload analyses. Additionally, the licensee will review the UFSAR to ensure the accuracy of the FPCC system design description. These issues will remain as an inspector follow item **(IFI 96-02-02)** pending NRC review of the UFSAR changes.

4.2 Quality Assurance in Engineering Activities

4.2.1 Self Assessment Activities

(a) Inspection Scope (37550)

The inspector reviewed the Nuclear Engineering Services Department (NESD) Procedure 18.01, Self Assessment Program, and reports of recent self assessments of the NESD covering performance, operating experience and the Field Revision Notice system. The NRC inspectors also reviewed a self assessment of the Nuclear Organization's corrective action program performed by a multi-disciplinary group, including the Quality Assurance Department.

(b) Observations and Findings

The report of the self assessment of the Nuclear Organization's Corrective Action Program, conducted December 11-21, 1995, reflected a critical review of the overall nuclear organization's corrective action program. The findings of the self assessment related to the NESD included numerous observations regarding backlogs in closeout of problem reports, review of vendor manuals, and operating experience reviews. PR 95.0661 was issued to follow-up on these items. The documentation closing this problem report included a thorough evaluation of the causes for the findings.

With respect to problem reports, while the inspectors found that a large number of items still remain open, the overall numbers have been reduced, even with an apparent lowered threshold for introduction of new items. The number of open problem reports has decreased from over 800 to about 400 during the past nine months. Similarly, the number of overdue items has decreased from about 90 to 10 during the same period. The reduction in these open items was attributed to management oversight, assisted by information provided to them by means of the Problem Report Monthly Report, a report that provides both numerical data and trending information.

With respect to vendor manuals, this matter had been identified by the licensee in 1994 and corrective action initiated. The impact of the problem and the compensatory measures implemented with respect to safety related work is discussed in depth in NRC Inspection Report 95-22. Increased management attention has been instrumental in reducing this backlog. Records indicate about 460 vendor manual-related open action items, including more than 180 overdue, in December 1995, have been reduced to 214 items, including 2 overdue, by late February 1996.

The self assessment of the operating experience program was initiated primarily in response to the non-compliance cited in Violation 95-14-01 related to improper disposition of General Electric Service Information Letter (SIL) 155 regarding cam follower cracking in SBM switches. This SIL had been issued originally in March 1976 with supplements issued in July 1976 and November 1979. The licensee's assessment indicated that this SIL was dispositioned in December 1986, along with more than 100 other operating experience reports dispositioned as part of a "backlog" project. NRC Inspection 95-14 found that BECo's 1980 response to the SIL and supplements had been inadequate as evidenced by subsequent SBM switch failures. This assessment included a review of 25 percent of the items closed in the 1986 effort and found no other improperly dispositioned items in that effort. The report of the assessment included five recommendations for improving the operating experience program, two of which were completed prior to the inspection with the remainder scheduled for completion by April 1996. One of the more significant recommendations was to include the status of the operating experience program's open items in the integrated action data base (IADB) which provides managers with status on all open corrective action items. A second recommendation, already realized as a side benefit of a reorganization combining the Technical Programs Division with the Operations Support Team, has expedited the screening and evaluation process because both functions are now performed by the same organization. While the backlog of open operating experience items has been reduced in the recent months, as of late January 1996, 81 items still remained open.

The self assessments of the Field Revision Notice (FRN) activities focused primarily on the cause or need for the field revisions and presented a basis for future trending. As such, the inspectors were unable to reach any conclusions regarding their effectiveness. One recommendation regarding the development of a set of criteria would allow the field revision of a plant design change without the initiation of an FRN. One specific criterion was that the intent of the modification or its safety evaluation not be changed. The inspector viewed this as a positive engineering initiative.

4.2.2 Problem Report Program

(a) <u>Inspection Scope (37550)</u>

The inspectors reviewed Nuclear Organization Procedure NOP92A1, Problem Report Program, ten problem reports generated within that system, and the follow-up and, where applicable, close-out documentation associated with these reported problems. The inspectors also interviewed the engineer responsible for the closure of the problem report related to the SBM switch problem which contributed to Violation 95-14-01. The NRC inspectors also interviewed the staff member responsible for the management of the computer-based IADB program, which was used for tracking the status of practically all open action items, including problem reports.

(b) Observations and Findings

The Problem Report program was structured to identify, classify/prioritize items by significance, defining action to be taken, establishing completion dates, and tracking items to completion. Reviews of records indicated that appropriate personnel had been trained (i.e., in the PR program, the specific root cause training area is evaluated in NRC Inspection Report No. 50-293/96-80) in the administrative requirements of the PR program procedure.

Problem reports that were reviewed by the inspectors identified a broad range of material and administrative problems. Reports were reviewed promptly by the Problem Assessment Committee (PAC) with appropriate actions and completion dates assigned. Reports which were closed were generally complete. Where required, root cause analyses or safety evaluations were usually thorough and of sufficient technical depth to provide a basis for the conclusions reached.

The root cause analysis performed for the SBM switch problem, documented in PR 95.9268, was extremely thorough and provided the basis for a well planned switch replacement program, Design Change 95-02. Since a small fraction of the affected switches require plant outage conditions for replacement, the replacement of these is not expected to be completed until March 1997, the next scheduled refueling outage. Contingency plans provide for earlier completion in the event of a forced outage.

During the closeout of the SBM switch problem, another defect was identified dealing with improperly peened rivets on which the switch cam followers rotate. The analysis of this new problem concluded that the switches would remain operable since the rivets were held captive in the switch. The switches affected by this latest problem are being replaced in the replacement program.

The inspectors reviewed several of the reports and supporting documentation provided from the IADB of which problem reports are just one element. The inspectors noted that this data base was being expanded with respect to the depth of information recorded which provided more readily retrievable data for programs such as the operating experience program. The inspectors also noted that the reports from the data base provided management with information to assist in focusing on the more critical issues while still providing trending and timeliness data to assist in their overall management efforts.

4.2.3 Quality Assurance Department Audits of Engineering Activities

(a) Inspection Scope (37550)

The inspectors reviewed the reports of two Quality Assurance Department audits, Audit Report 95-06, Special Nuclear Material and Refueling Activities, and Audit Report 95-07, Core Spray System. The inspectors also reviewed

Quality Assurance Surveillance Reports directly or indirectly related to Engineering associated activities.

(b) Observations and Findings

The audit reports reflected objective, in-depth examination of activities important to safety. Problem Reports were initiated to document and provide for appropriate review and follow-up of items assessed as unsatisfactory; recommendations were tracked as recommendation reports. Recommendations were technically sound and appeared to be enhancements to safety.

(c) Conclusions on Quality Assurance in Engineering Activities

Overall, the inspectors concluded that the licensee has programs and administrative controls in place that provide the basis for corrective actions. Recent self assessments conducted by the NESD and QA have been effective in identifying problems; close out of problem reports stemming from self assessments have been effective in determining root causes. Problems are identified at various organizational levels and promptly reviewed for impact at appropriate operational and safety committee levels. The IADB provides a useful single source of status information for cognizant management. A significant number of items reported in this system existed and licensee estimates of 1100 to 1200 new items per year can be expected to keep the total numbers at or above present levels.

4.3 Miscellaneous Engineering Issues (92702)

(a) (Closed) Violation 50-293/95-14-01

Significant conditions adverse to quality were not promptly corrected. Specifically, in November 1994, the licensee identified significant roof leakage into the Emergency Diesel Generator Building near safety-related electrical equipment and corrective action was not taken to repair the roof. Secondly, a safety-related type SBM control switch for high pressure coolant injection (HPCI) system valve 2301-8 failed to operate in November 1992 and corrective actions were not taken to preclude repetition such that, in May 1995, the control valve for HPCI torus suction valve 2301-35 failed to operate due to cracked lexan cam followers, the same cause as that of the November 1992 failure.

The inspectors reviewed the licensee's corrective action as summarized in their letter, dated August 18, 1995, in response to the Notice of Violation (NOV) and as detailed in internal documents including the following:

RC95.0036.02 (Problem Report and Operability Evaluation Refresher Training)
RC95.0036.03 (Review Older Open PRs Periodically)
RC95.0036.04 (OERP Program Self-Assessment)
RC95.0036.05 (Replace EDG Roof)
RC95.0036.06 (Replace Susceptible SBM Switches)

The latter two items address actions related to resolution of the specific problems; the first three address corrective action with respect to their root causes. Based on review of various problem reports, the completed corrective action, and the overall trend of problems identified, reviewed, and resolved, the inspectors concluded that the corrective actions were satisfactory.

5.0 PLANT SUPPORT (71750)

5.1 Silica Concentration in Spent Fuel Pool

On March 20, 1996, the chemistry department issued PR 96.9122 to document silica concentration in the spent fuel pool increasing at an unexplained rate. Since the degradation of the spent fuel pool Boraflex racks was identified as a potential cause for this increased silica concentration, the inspector discussed this issue with Chemistry Department management to determine the significance of this data and adequacy of actions performed and/or planned.

Industry experience has shown that Boraflex panels similar to that installed in some of the spent fuel pool racks at PNPS have demonstrated shrinkage which is manifested by stress-induced gaps in the Boraflex sheets (reference NRC Information Notices 93-70 and 95-38). Boraflex is a silicon-based polymer that includes an embedded Boron-10 poison to assure that a shutdown margin is maintained in the spent fuel pool due to the storage of spent and new fuel assemblies. Therefore, increased levels of silica in the spent fuel pool water may indicate a degradation in the Boraflex.

No water quality limits exist for silica in the fuel pool at PNPS; therefore, no operational actions were required in response to the elevated concentrations. As an initial response to the increased silica levels, BECo secured the fuel pool cleanup demineralizer system since analytical results indicated silica removal exhaustion of the system. The demineralizer was later put back in service to maintain pool water clarity to support work in the pool. In addition, a meeting between engineering, operations and chemistry was held which reviewed the data and confirmed that an immediate concern of the integrity for the Boraflex racks did not exist.

The inspector discussed this information with chemistry management and confirmed that they have discussed their data with other licensees who have experience with Boraflex degradation. Current data indicates that immediate action is not required. However, the licensee planned to continue to retrieve and analyze historical plant data. In addition, the licensee planned to send plant personnel to other utilities to observe different Boraflex inspection methods (i.e. aerial density and blackness tests) in order to gain experience with these methods and information that may be applied to PNPS.

The inspector concluded that BECO responded well to the higher indications of silica in the spent fuel pool. Personnel communicated with other licensees to obtain industry information and place BECo's data in perspective. When site visits and historical data review has been completed, further recommendations will be considered. Since no limits exist for silica levels in the spent fuel pool water and BECo continues to evaluate industry operating experience, no immediate safety significant concern exists. BECO's continued evaluation of this issue was appropriate. The inspector also discussed this issue with the NRR Project Manager who indicated that the NRC staff has the issue of boraflex degradation in spent fuel pool racks under further review.

5.2 Radiation Controls

5.2.1 Organization Changes

Since the last radiation controls inspection, the licensee has eliminated a layer of plant management, which affected the radiation protection (RP) organization. The previous Radiological Operations Division Manager was assigned supervision of the As Low As Is Reasonably Achievable (ALARA) group, and reports directly to the Radiation Protection Manager (RPM). The inspector viewed the additional emphasis provided to ALARA and the RPM's closer relationship to the radiological operations as an improvement in the RP organization. Another affect of the licensee's reorganization has been the elimination of four RP technician positions (from 29 down to 25 currently). During this inspection, a shortage of RP technicians was noted in the RP instrument issue area. The inspector observed RP technicians were in need of survey instruments and none were available, and there were no RP technicians available to source check and supply additional instruments. The apparent effect of this situation was delayed RP support for radiological work. The Radiation Protection Manager indicated that at present, the department was down to only 22 RP technicians and three additional RP technicians were being hired. Given the above shortage, the inspector identified no adverse effect on safety performance.

5.2.2 Quality Assurance

The inspector reviewed the licensee's quality assurance (QA) assessment of the RP program effectiveness. The latest RP audit (No. 95-04) was conducted during March through May 1995. The QA audit was conducted during the 1995 refueling outage and was of good depth and appropriate scope. The inspector noted that no outside technical reviewer was included in the audit team. In addition to an annual comprehensive RP program review, three 1995 QA surveillances and two 1996 QA surveillances were performed to review aspects of the RP program. The areas chosen for review were safety significant program areas and were of a reduced number than the previous 2 years. The Quality Assurance Manager indicated that the RP department was moving toward a self-assessment surveillance approach, although not currently in effect. The inspector noted no decline in RP program performance during this inspection and therefore, the change in QA surveillance did not appear to have any near-term negative effects.

5.2.3 Radiological Problem Report Program Review

The inspector selected and reviewed 26 of the more radiologically significant radiological problem reports (RPRs) written since 1995. In general, the radiological incidents have been fewer in number and of less safety significance than during the previous 2 years. The threshold for reporting radiological incidents continues to be low, providing an excellent source of feedback on station radiological performance. The resolution of the RPRs continues to be generally good. The inspector did observe several examples where radiological incident causes were not completely or accurately determined and where other causes and more effective corrective actions could have been pursued. For example, problem report (PR) 95.311 documented an incident where a worker entered a high radiation area (the drywell) without a thermoluminescent dosimeter (TLD), having inappropriately given his TLD along with his security badge to a security guard stationed at the drywell entrance to log entry and exit from this vital area of the plant. The RPR evaluation determined the cause of the incident to be inattention-to-detail; however, the reason for the TLD to be removed from the security badge and the security badge exchange process itself was not evaluated. Due to other plant process reviews, the licensee has since eliminated the need for a security guard at the drywell entrance, but the RPR should have identified this cause and eliminated this requirement as part of the problem report resolution process. Another problem report, PR 95.429, documented an exposure concern questioning the need for three individuals to perform control rod drive replacement when two individuals had successfully performed the task on one occasion. The problem report dispositioned the concern based on a maintenance supervisor's statement that three individuals were necessary. No reevaluation of the work task requirements based on two individuals was performed.

In summary, the radiological problem report process continues to provide an excellent identification of radiological incidents; however, in some cases, critical evaluation of root cause and subsequent assignment of corrective actions was lacking.

5.2.4 Radiation Protection Instrument and Dosimetry Calibration

The inspector reviewed the various facilities used for radiation instrument and dosimetry calibrations, reviewed calibration records, and interviewed cognizant licensee personnel.

RP procedures specify semi-annual calibration of radiation protection instruments and annual calibration of health physics (HP) counting instruments and whole body counter calibrations . The inspector reviewed instruments available for use and determined that the calibration frequencies had been met.

The inspector reviewed the basis for instrument calibrations through discussions with the Supervisor of Radiological Instruments and demonstrations in the instrument calibration facility. The inspector observed the methodology used to calibrate the instrument calibration sources on an annual basis with traceability to the National Institute of Standards Technology (NIST). The inspector determined that the NIST calibrated electrometer was used to verify source calibration at one value with an acceptance criterion of $\pm$ 5% from an expected decay-corrected value. With the acceptance criterion met, the original source calibration data was decay-corrected and source strength values were specified for each source attenuator position. The TechOps 50 Curie cesium-137 source was recently compared to a NIST traceable electrometer indicating 3% higher than the calculated value. The TechOps source was originally calibrated in 1972 and the original data were used to determine current source strength for all 24 source strength values. Actual electrometer measured values for specifying source strengths at each point, to be used for instrument calibrations, were not utilized. The one value that was compared to the NIST traceable electrometer was not used to replace the 1972 data value. The source calibration procedure (No. 6.6-113, Revision 3) does not describe the derivation of source strength values to be used for instrument calibrations. The Radiation Protection Manager stated that the procedure would be further developed, at a minimum, to document the current source calibration practice and to evaluate calibration method improvements as necessary. The inspector determined that the requirement for an annual NIST traceable calibration had been met and no discrepancies with the regulations were noted.

The inspector reviewed the HP counting laboratory that was used for air sample analyses to evaluate the adequacy of high purity germanium (HpGe) detector calibration and operation. The inspector compared current 10 CFR 61 waste stream analysis information with the radionuclide analysis library used to identify radionuclides contained in air samples and determined that a very extensive library of radionuclides had been developed. The inspector reviewed documentation indicating that the HpGe detectors had been calibrated on April 1, 1995, utilizing NIST traceable sources for various counting geometries. The calibration included determination of quality control charts for comparing daily counting performance comparisons to a known value. Quality control charts were utilized based on a ±5% and ±10% from a known value. Due to the statistical basis of radioactive decay, normal quality control charts are plotted based on standard deviations away from a known value. When questioned by the inspector, the Respiratory Protection Supervisor indicated that the standard deviation values were found to be too wide and therefore, chose more restrictive values for indicating reliable instrument performance. The inspector observed the sample positioning arrangement on top of the detector end cap and noted that two cylindrical shelves of different heights were used to count particulate filter and charcoal cartridge air samples. The cylindrical shelves "floated" on top of the detector end caps without a fixed orientation. The particulate filter sample placement on top of the shelves also allowed for a non-fixed orientation. The inspector determined that the "floating" orientation of the sample to the detector could cause the broad statistical variation. The Supervisor of Respiratory Protection stated that the HpGe detector counting orientation would be fixed. The inspector determined that there was very low safety consequence due to this finding.

The inspector reviewed calibration and daily source check documentation for the two whole body counters used for bioassay measurements of personnel. The last annual calibrations were done on June 29, 1995, and July 3, 1995, using a NIST-traceable 8-isotope source. Appropriate quality control charts were developed and were used for daily source count checks on system performance. No discrepancies were noted with the calibration or daily use of the whole body counters.

The licensee utilized a vendor, Yankee Atomic Electric Company, for processing TLDs for determining record external exposures for Pilgrim's occupationally-exposed radiation workers. The inspector reviewed the most recent National Voluntary Laboratory Accreditation Program (NVLAP) TLD testing and inspection results with respect to regulatory requirements. The Department of Commerce certified Yankee Atomic Electric Company NVLAP accreditation until September 30, 1996. The inspector's review of the TLD testing indicated very good performance results in all eight testing categories as defined in ANSI-N13.11-1983. The NVLAP onsite assessment report dated April 21, 1995, required several quality improvement deficiencies be corrected and the inspector reviewed timely closeout correspondence of the deficiencies. No discrepancies with TLD calibration/processing were noted.

The inspector observed the methodology for calibration of electronic dosimeters (EDs). Upon entering the Turbine Building 23-foot elevation calibration room, the inspector identified that the vertical source actuator rod that allows a 15 millicurie cesium-137 source to be withdrawn, was not locked. There was a locked padlock attached to a short chain; however, the chain was not fastened over the source actuator rod. The inspector determined that the exposed source resulted in a 60 mrem/hr exposure at a distance of 30 centimeters from the source. The Senior Radiological Engineer called an RP technician to immediately post and barricade the area, while another RP technician was dispatched to obtain a key and the source was subsequently locked. The inspector noted that the calibration room was normally kept locked and only RP personnel have access to a key to this facility. However, the primary method for source control was the source locking mechanism that was not secured.

Procedure No. 6.6-010, Revision 6, "Inventory, Control and Leak Checking of Radioactive Test Sources," Section 8.6 states "When using calibration sources greater than or equal to Tech Spec limits (i.e., 100 uCi of Beta and/or Gamma,..., the following practices shall be followed; .... (d) Ensure the source is locked and the keys controlled when the source is not in use." Technical Specifications, Appendix A, Section 6.11, requires that procedures for personnel radiation protection shall be adhered to for all operations involving personnel radiation exposure. Failure to adhere to procedure No. 6.6-010 was a violation **(VIO 50-293/96-02-03)**.

After a demonstration of the electronic dosimeter calibration methodology and examination of selected electronic dosimeters, the inspector determined that the semi-annual calibration of electronic dosimeters was found to be in accordance with regulatory requirements.

In summary, the calibration of RP instrumentation was found to be generally very good, with procedure enhancements identified for source calibration and standardization of HpGe detector counting geometry. A violation was identified due to failure to lock a calibration source as required.

5.2.5 External Exposure Control

The inspector toured the major areas of the RCA, interviewed workers and HP technicians, and made independent radiation field measurements. The inspector observed that the licensee had opened up significant areas of the plant as clean areas that were historically contaminated areas. For example, the reactor building corner rooms, the control rod drive rebuild room, and the residual heat removal (RHR) valve rooms were all reclaimed territory accessible now without the need for protective clothing. Postings, barricades, and locked areas were found to be in accordance with regulatory requirements except for the calibration source mentioned in the previous section of this report. Due to a recent change in security access to the protected area via electronic palm readers, the security badges and attached TLDs are no longer issued by security personnel and the badges are available to the general workforce. The Radiological Department had appropriately responded to ensure that the correct TLD is picked up and worn by the worker by requiring a bar-code read of the TLD during entry to the RCA. The inspector determined this to be a very good improvement to the personnel exposure monitoring program. The inspector observed very good administration of RWP briefings and response to personnel and equipment release from the RCA. Future changes include the reconstruction of the "red line" (primary access point to the radiological controlled area (RCA)) and the RCA tool supply depot. At the time of this inspection, these future changes were still in the design stage.

5.2.6 Exposure Reduction

During 1995, Pilgrim Station reported 410 person-rem during the 1995 refueling outage versus a goal of 270 person-rem, and a 1995 annual exposure total of 482 person-rem versus an annual goal of 380 person-rem. By reference to NUREG-0713, Volume 16, Table 4.5, Pilgrim Station's exposure performance with respect to other BWRs has ranked below the average BWR over the 5-year period of 1990-1994. Over the same time period, the industry BWRs have seen an average decline in annual exposures of approximately 5% per year. The 1994 average BWR exposure was 327 person-rem, therefore, Pilgrim's 1995 exposure of 482 person-rem continues to indicate below average BWR exposure performance. With respect to this performance, the inspector reviewed the exposure reduction efforts being taken by the licensee to improve overall exposures at the station.

The inspector reviewed the licensee's refueling outage (RFO) 10 ALARA report to investigate the reasons for relatively poor exposure performance during the 1995 refueling outage. According to the licensee evaluation, there was over 30 person-rem attributed to unplanned emergent work activities, and schedule overruns for the scheduled work activities comprised the bulk of the additional exposure overruns. According to the licensee's outage ALARA report, corrective actions indicated a need for more advance time for planning, increasing the use of remote monitoring, and providing ALARA incentives/penalties in vendor contracts.

The inspector reviewed an outage modification that resulted in over 22 person-rem over budget; reactor water cleanup isolation valve replacements, motor-operated valves 1201-2 and 1201-5. The valve replacement was in the original scope of refueling outage No. 10 (RFO10) that occurred from April to June 1995. The valves were obtained from Europe and were a long lead time item. The detailed maintenance planning was delayed until documentation from the manufacturer was obtained. The ALARA review process was contingent on completion of the detailed maintenance planning. Based on an interview with the ALARA specialist, in this instance, the plant design change (maintenance plan) was not ready until approximately 6 weeks prior to the start of the outage, at which time the exposure reduction plans were initiated.

Hydrolasing of the RWCU suction line was prescribed and performed. The hydrolase was prematurely terminated and later radioactive crud resettled into the piping near these valves. Therefore, there was no net reduction in dose rates. Also, during performance of the valve replacement activities, one of the valves was installed backwards, improper welds required rework, and the wrong valve packing was installed and replaced twice. During removal of the 1201-2 valve from the drywell, the 8 R/hr valve arrived at the drywell step-off pad with no prearranged plan for disposal. Temporary shielding was placed on the valve at that point and a radwaste disposal plan was determined. The valve replacement project was completed resulting in 44.86 person-rem versus the 22 person-rem estimated. Although short lead time planning was the main cause suggested in the outage ALARA report, the inspector determined that lack of effective management oversight of the work was another primary cause of the poor exposure performance of this work activity.

Since RFO10, the licensee has not had many opportunities to demonstrate improved planning and work control performance. In January 1996, a 1-inch drain line was removed from underneath the RWCU regenerative heat exchanger. For this minor modification, very effective exposure reduction techniques and excellent work performance was demonstrated resulting in 0.7 person-rem versus 1.0 person-rem as estimated. For this job, maintenance planning began 5 weeks in advance of the work, and good coordination between the various work groups was achieved. Additionally, the Radiation Protection Department piloted a Radiation Protection Work Practices Team effort with the emphasis on integration of the work control process to include worker input and RWP development, and geared to develop worker ownership in the work process. Based on the limited application of this approach, its effectiveness appears to be a positive control measure.

In a previous inspection[1], the civil/structural engineering group committed to revising the shielding procedure, NEDWI 239, Rev. 6, "Criteria For Evaluation And Approval Of Radiation Shielding." The new revision was to incorporate a generic scaffold-loaded shield design and various previously approved specific temporary shielding applications. During this inspection, the inspector determined that no procedure revision has yet been made. The Radiation Protection group indicated that the engineering procedure, NEDWI 239, was being tracked on the Dose Reduction Action Plan (DRAP) with a target due date of January 1997.

The DRAP was administered by the Radiological Technical Support Team and consisted of approximately 85 open items. The DRAP is derived from licensee-generated ALARA reports, station department generated ALARA action plans, the Institute for Nuclear Power Operations, Nuclear Regulatory Commission, and employee suggestions. This appears to be a good program for recommending exposure reduction initiatives. Although the DRAP items are station-wide, the assignment of action items outside of the RP organization was done on a cooperative/voluntary basis by the Radiological Technical Support Team Leader. When a DRAP item misses a due date, the item is included as an agenda item for

[1] Inspection No. 50-293/94-15, conducted in July 1994

the monthly ALARA Committee meeting. The inspector noted that not until items become overdue does station management typically get involved in the exposure reduction items. Management-directed work assignment and accountability through the line organization is not currently utilized.

The inspector investigated how ALARA decisions were made and discovered that in general, they were made at the staff level in two RP groups utilizing two different cost versus benefit criterion. In the Radiological Technical Support Team, a Senior Radiological Engineer had developed a cost/benefit method that estimated labor costs per pound of lead blankets to install and remove them for four areas of the plant. This method assumes a simple hand-carried approach and does not take into account large load lifts of shielding packages into locations. Also, this method does not take into consideration a common labor pool that is typically available for an outage and may not result in additional labor costs. Exposure reduction decisions made by the Radiological Technical Support Team influence the outcome of DRAP item recommendations.

In the ALARA Programs Team, an ALARA specialist utilized a different approach at ALARA decision making. Based on the reviews of scheduled maintenance work packages, the ALARA specialist estimated the exposure value with and without shielding and estimated the exposure cost to install and remove temporary shielding. The cost/benefit analysis was based on net exposure reduction and was often limited by engineering analysis of maximum pipe loading capacity. Some of the limitations of this approach were the hand-carrying of lead blankets assumed in exposure cost calculations and only direct pipe loading of temporary shielding was considered in the shield design concept. Another limitation was the work package scope of the ALARA consideration. Although the ALARA Program Team has attempted a comprehensive outage area exposure view, in some cases, the accuracy of these efforts and scope of review continues to be limiting.

In summary, the inspector determined that the Radiological Department utilized different methodologies for making exposure reduction decisions and that these decisions are often made at the staff level in the organization with limited station-wide focus and management involvement concerning exposure reduction decisions.

When interviewing Civil/Structural Engineering Group personnel, the inspector learned of plans to move forward with the permanent shielding of large system components (e.g., the recirculation and RHR piping systems). While still in the planning stage, the inspector noted that the basis for permanent shielding of the drywell was the RFO10 temporary shielding. The permanent shielding plan, as proposed at the time of this inspection, suggests that the net reduction in exposure would be approximately 10 person-rem in subsequent outages due to the savings from not requiring installation or removal of the shielding. Notwithstanding, the application of this drywell shielding plan would not be expected to greatly alter Pilgrim Station's below average exposure ranking with other BWRs.

5.2.7 Internal Exposure

The inspector observed very good contamination control techniques and effective air sampling practices during tours of the station. Utilizing data supplied by the licensee, DAC-hour tracking (as determined by air sampling) was compared with DAC-hours assigned via whole body counting. The data indicated five times as many internal exposures were determined through whole body counts than from work area air measurement determinations. The levels were below recording requirements; however, this may be indicative of less than adequate air sampling practices. The Respiratory Protection Supervisor and the Radiation Protection Manager (RPM) were advised of this information and the RPM stated that implementation of the air sampling program would be reviewed.

5.2.8 Radwaste Equipment Material Condition Review

In response to a recent plant condition observed in a Region I nuclear plant involving degraded radwaste equipment (as documented in NRC Information Notice 96-14), the inspector reviewed the Pilgrim Station radwaste facilities to verify the physical condition of nonroutinely-visited areas. The inspector reviewed radwaste building floor elevation blueprints and discussed with the Radwaste System Engineer the areas of interest. The inspector accompanied by the Radwaste System Engineer toured the areas listed below.

> Radwaste Building minus-1-foot elevation:
> cement solidification bin (no liquid contents), waste feed tank (contents indeterminate based on outside tank observation)

Due to dose rate considerations, the Radiation Protection Department arranged for remote surveillance by video camera of the radwaste concentrator room and attempted remote surveillance of the floc recycle tank.

> Radwaste concentrator room (partially dismantled in 1986): The room was devoid of liquid contents although the room was quite cluttered with debris. Floc recycle tank (abandoned in 1978): the tank did not show any signs of leakage or tank degradation, however, the physical contents could not be ascertained. A contact dose rate of 30 R/hr indicated some radioactive contents existed, however, determination of liquid or solid phase needs to be determined.

In all cases, the tanks, valves, and piping systems were found to be leak-free, with no detectable corrosion or degradation of plant components. To prevent future degradation of components, several tanks were identified (as mentioned above) that either contained liquid contents or the tank level could not be determined based on a visual inspection outside of the tank. The Radwaste System Engineer stated that the indicated tanks would be evaluated to determine contents and that a disposition plan for those that contain liquid contents would be developed.

5.2.9 Radwaste Procession

On March 26, 1996, the inspector was notified by telephone by the Radiation Protection Manager, of a March 25, 1996 event involving the inadvertent release of a radioactive material labeled bag of chemical laboratory trash to a battery recycling facility in Woburn, Massachusetts. The battery recycling facility was not licensed to receive radioactive material and upon finding the yellow radioactive material bag within a shipment of batteries, immediately notified BECo. BECo RP technicians were dispatched on March 25, 1996, to survey and reclaim any licensed radioactive material. Approximately 2000 disintegrations per minute of activity was measured from inside the bag and no measurable smearable contamination was detected outside the bag, in the shipping container, or anywhere on the battery facility's premises.

The radioactive material trash bag was brought back to Pilgrim Station and was further examined. The bag was marked December 18, 1995 and contained chemical laboratory waste paper and disposable vials originating from the Pilgrim radiochemical laboratory. Further RP technician measurements determined the bag to contain approximately 14 nCi of radioactive material. The licensee initiated a level one problem report and made a preliminary determination that the radioactive material bag had been mishandled during waste collection and sorting activities. Immediate corrective action taken by the Radwaste Department involved making separate clean waste and radioactive waste pickups inside the plant. The inspector was notified of the above information by the Radiation Protection Manager on March 26, 1996. The inspector determined that the licensee had failed to control licensed material and that this was a violation of 10 CFR 20.1802 **(VIO 50-293/96-02-04)**. The inspector notified the RPM by telephone on March 27, 1996 that a violation of 10 CFR 20.1802 had occurred. The RPM acknowledged this finding.

5.2.10 Review of Updated Final Safety Analysis Report Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description.

While performing the inspections discussed in this report, the inspector reviewed the applicable portions of the UFSAR that related to the areas inspected. The following inconsistency was noted between the wording of the UFSAR and the plant practices, procedures and/or parameters observed by the inspector. The inspector reviewed the implementation of the Pilgrim radiation control program with respect to Section 12 of the Pilgrim FSAR. In general, the Pilgrim radiation control program is implemented in accordance with Section 12 of the UFSAR. One minor inconsistency was noted with respect to Section 12.4.2.2 where the use of electronic dosimeters is not mentioned. No other inconsistencies were noted.

6.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (40500, 92700)

6.1 (Closed) LER 96-01: Torus-To-Reactor Building Vacuum Relief System

The inspector reviewed Licensee Event Report (LER) 96-01 submitted to the NRC to verify accuracy , description of cause, previous similar occurrences, and effectiveness of corrective actions. The inspectors considered the need for further information, whether the event warranted further onsite followup. The LER was also reviewed with respect to the requirements of 10 CFR 50.73 and the guidance provided in NUREG 1022 and its supplements.

LER 96-01, dated April 4, 1996 reported a licensee discovery of an inoperable differential pressure switch designed to actuate on a low pressure condition in the torus relative to the reactor building. This condition would not have allowed operation of one of the two torus-to-reactor building relief valves to automatically open on a sufficient differential pressure condition. The inspectors' review of this condition and BECo's subsequent actions is described in Section 3.2 of this report. The inspector determined that the LER was complete and the licensee's corrective actions for this problem were timely and appropriate. This LER is closed.

6.2 Procedure Quality and Usage Issues

As a result of the procedural usage and adherence issues identified in NRC Inspection Report 50-293/96-80, the inspector reviewed the collective findings concerning procedure usage and adherence during this routine inspection period. Some pertinent examples, of both positive and opportunities for improvement, were noted:

- Section 2.2 involves a licensed operator strictly following an incorrect procedure causing a recirculation motor-generator set scoop tube lock-up.
- Section 2.4 notes that the tagging program was well controlled and implemented.
- Sections 3.2 and 3.3 notes good maintenance worker procedural adherence during two separate troubleshooting activities.
- Section 3.3 notes that QA identified discrepancies in electrical torque specification selection as a procedural adherence and procedure clarity issue.
- Section 3.4 discusses the use of an outdated maintenance procedure during the rebuild of the "A" recirculation pump mechanical seal (Circa RF010). Also, the UFSAR description of the seal life needs to be updated to reflect the newer design.
- Section 3.6.3 notes that the work control program requirements for the preparations of maintenance work instructions were not clearly defined to state managements expectations in this area.

- Section 5.2.3 identified that a procedure violation occurred concerning an instance of inadequate controls for locking a radioactive calibration source.

The inspector concluded that there was some evidence of procedure quality and usage issues being identified, corrected and trained upon during this period. Opportunity remains to minimize the potential for future consequential problems such as the scoop tube lock-up event that occurred during this period.

7.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES

7.1 Routine Meetings

Two resident inspectors were assigned to Pilgrim Nuclear Power Station throughout the period. Backshift inspections were performed on March 6, 13, 22, 28 and April 1 and 4 and deep backshift inspections were performed on March 16 and 23. On March 25 and 26, Mr. J. Wiggins, Director, NRC Region I Division of Reactor Safety, visited the site. On March 28 Mr. R. Cooper, Director, NRC Region I Division of Reactor Projects, and Ms. S. Shankman, Acting Director of the Office of Nuclear Reactor Regulation Division of Reactor Projects Project Directorate I-1 visited the site. Messrs. Cooper and Wiggins and Ms. Shankman visited the site to interview senior and plant level management, tour the plant, and hold discussions with the resident inspectors.

At periodic intervals during this inspection, meetings were held with senior BECo plant management to discuss licensee activities and areas of concern to the inspectors. Each specialist inspector held an inspection debrief at the completion of their inspection. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting on May 3, 1996 summarizing the preliminary findings of this inspection. No proprietary information was identified as being included in the report.

7.2 Other NRC Activities

From February 26 through March 8, 1996, Mr. J. Noggle performed a routine radiological protection and external exposure inspection. From March 11 through 15, Messrs. D. Limroth and S. Chaudhary performed a routine engineering inspection. Also, from March 11 through 15 Mr. J. Caruso performed an LCO maintenance initiative inspection. Also, the NRR Project Manager, Mr. R. Eaton, conducted a review of the spent fuel pool licensing and design bases. The results of these inspections have been integrated into this report.

From March 18 through March 22, 1996, Mr. R. Albert performed a routine security inspection. The results of this inspection are documented in NRC Inspection Report 50-293/96-04.

7.3 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 UFSAR commitments. For an indeterminate amount of time, all reactor inspections will provide additional attention to UFSAR commitments and their incorporation into plant practices and procedures. While performing inspections which are discussed in this report,the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The following inconsistencies were noticed between the wording of the UFSAR and plant practices or procedures and/or parameter observed by the inspector.

- Report Section 3.4, UFSAR Section 4.3, the UFSAR wording refers to a recirculation pump seal cartridge expected life of 1 year with a 90% probability factor. In actuality, the newer seal designs has an expected life of 4 years which is more consistent with the 2 year operating cycle.

- Report Section 4.1, UFSAR Section 4.8 and 10.4, the SFP limiting parameters are not included in the description and various other minor related updates tracked as NRC IFI 96-02-02.

- Report Section 5.2.10, UFSAR Section 12.4.2.2, the wording does not include the use of electronic dosimeters.