U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	50-293
Report No.:	50-293/92-14
Licensee:	Boston Edison Company 800 Boylston Street Boston, Massachusetts 02199
Facility:	Pilgrim Nuclear Power Station
Location:	Plymouth, Massachusetts
Dates:	June 16 - July 27, 1992
Inspectors:	J. Macdonald, Senior Resident Inspector A. Cerne, Resident Inspector D. Kern, Resident Inspector

Approved by: Richard L. Emch, Jr. 8/7/92
for E. Kelly, Chief, Reactor Projects Section 3A Date

Inspection Summary:

Areas Inspected: Routine safety inspection of plant operations, radiological controls, maintenance and surveillance emergency preparedness, security, safety assessment and quality verification, and engineering and technical support.

Results: Inspection results are summarized in the Executive Summary.

Violation: One violation was identified for the failure of the licensee to properly implement corrective action programs as required by 10 CFR 50 Appendix B, Criterion XVI with respect to an unauthorized non-code repair to the salt service water system (VIO 50-293/92-14-01, Section 4.2)

EXECUTIVE SUMMARY

Pilgrim Inspection Report 50-293/92-14

Plant Operations: A weakness was identified in the implementation of procedural instruction to ensure comprehensive review of evolutions with the ability to effect reactivity such as by causing inadvertent recirculation pump trips. Subsequent response to this weakness was effective. Introduction of robotic condenser bay tours is a positive initiative. Evaluation and continuing monitoring of a slight reactor water cleanup system heat exchanger leak has been appropriate. Operator response to the July 26, "A" recirculation pump trip was well controlled.

Radiological Controls: Sludge liner transfer operations were well controlled. Appropriate supervisory and QA oversight were present. Workers were knowledgeable of established safety precautions and management concern for worker safety and dose savings were evident. A recent revision to the dosimetry issuance program has been effectively implemented.

Maintenance and Surveillance: Response to a stripped terminal point during reactor core isolation cooling system surveillance testing was appropriate. The in-progress testing was aborted and corrective maintenance was promptly completed minimizing safety system unavailability.

Inadequate response to the identification of an unauthorized salt service water system repair was noted. Station personnel initially failed to document the deficiency and repair and failed to notify appropriate levels of management. Additionally, the unauthorized repair failed to conform with ASME Class 3 code repair requirements. Finally, upon discovery of the unauthorized repair activity, appropriate corrective actions were not implemented in a timely fashion.

Emergency Preparedness: Station response to the June 24, medical emergency was well coordinated in accordance with established procedure. Station emergency medical technicians and the security force interaction with the Town of Plymouth ambulance ensured prompt care for the stricken individual.

Security: Strong security force performance was noted during extended Jeep backshift observation. Officers and guard force members were alert and responsive to assigned duties. Transfer of alarm station command during shift turnover was well executed.

Safety Assessment and Quality Verification: Phase I of a three phase organizational restructuring initiative was effectively implemented. A Technical Specification required special report documenting the inoperability of the diesel driven fire pump was submitted in a timely manner.

Engineering and Technical Support: The salt service water system pipe replacement project continued to progress in a controlled fashion. A conservative approach to the identification and resolution of nonconforming conditions was evident. The status of several significant engineering initiatives were reviewed and determined to be appropriately evaluated and prioritized.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

At the start of the report period, Pilgrim Nuclear Power Station was operating at approximately 100% of rated power. Throughout the report period, brief weekly reductions to 95% power were conducted to perform control rod exercise activities.

On June 24, the station requested the Town of Plymouth ambulance to assist a training instructor who had exhibited symptoms of cardiac distress while located in the control room. The instructor was transported via ambulance to Jordan Hospital for further treatment.

On July 18, reactor power was reduced to approximately 50% in order to conduct a thermal backwash of the main condenser. During the power reduction, the hydrogen water chemistry system was secured to reduce dose rates during a condenser bay tour and completion of miscellaneous minor maintenance activities. The reactor returned to 100% power on July 19, following the backwash. Additionally on June 18 and 19, the reactor core isolation cooling system was briefly declared inoperable to repair a stripped contact point on the leak detection subsystem terminal block.

On July 26, the "A" recirculation pump motor generator set tripped at 9:23 pm when its associated lockout relay actuated. Reactor power stabilized at approximately 60% power and all systems responded to the motor generator set trip as designed. The licensee concluded the lockout relay actuated due to minor pump speed oscillations experienced at greater than 96% of speed control. On July 27 at 2:23 am, the pump was restarted and the reactor returned to 100% power later on July 27 at 12:10 pm.

Additionally, the HPCI system was removed from service on July 27 for a short planned system outage to conduct miscellaneous maintenance activities including lubricating oil change, installation of diagnostic test sensors of motor operated valves, and control fuse replacement. The system was scheduled to be returned to service on July 29.

At the conclusion of the report period, Pilgrim Nuclear Power Station was operating at 100% rated power.

2.0 PLANT OPERATIONS (71707, 40500, 90712)

2.1 Plant Operations Review

The inspector observed plant operations during regular and backshift hours of the following areas:

Control Room	Fence Line (Protected Area)
Reactor Building	Turbine Buildi
Diesel Generator Building	Screen House
Switchgear Rooms	
Security Facilities	

Control room instruments were observed for correlation between channels, proper functioning, and conformance with Technical Specifications. Alarms received in the control room were reviewed and discussed with the operators. Operator awareness and response to these conditions were reviewed. Operators were found cognizant of board and plant conditions. Control room and shift manning were compared with Technical Specification requirements. Posting and control of radiation, contamination, and high radiation areas were inspected. Use of and compliance with radiation work permits and use of required personnel monitoring devices were checked. Plant housekeeping controls, including control of flammable and other hazardous materials, were observed. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout, and lifted lead and jumper logs. Inspections were performed on backshifts including June 16 - 19, 23 - 25, and 29 and 30, July 6, 8, 13, 14, 16, and 20 - 23, 1992. Deep backshift inspection was performed on June 18, 10:00 pm - June 19, 3:50 am, June 23, 10:00 pm - June 24, 3:30 am, June 24, 10:00 pm - June 25, 1:15 am, and June 25, 10:00 pm - 10:45 pm.

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

2.2 Procedural Guidance for On-line Recirculation Pump Motor Generator Brush Replacement

The inspector reviewed several procedures pertaining to on-line recirculation pump motor generator (MG) set brush replacement to verify agreement with Technical Specifications, incorporation of industry guidance, and clarity of instruction to operations personnel. Procedure 1.2.2, "Administrative Ops Requirements," promulgates several operational policies intended to enhance overall plant safety. One such provision in the procedure directs that planned maintenance activities which have the potential to trip a recirculation MG set shall be performed at or below the 80 percent control rod pattern line. The licensee had implemented this operational precaution in response to NRC Bulletin No. 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)." This bulletin requested licensees to take action to prevent the occurrence of power oscillations due to entry into the power to flow region of instability and to ensure that the safety limit for the plant minimum critical power ratio (MCPR) was not exceeded.

The inspector noted that the June 1, 1992 recirculation pump MG set brush replacement had been initiated while operating at the 100 percent control rod pattern line. As previously documented in NRC Inspection Report 50-293/92-08, review of core performance data following the inadvertent recirculation pump trip (RPT) verified that no thermal limits had been exceeded nor had the region of power to flow instability been entered. Therefore, initiation of this maintenance activity from above the 80 percent rod line had minimal safety significance. However,the inspector questioned if this activity could have resulted in entering the region of instability, had it been initiated from other reactor power to flow conditions above the 80 percent rod line which would be contrary to the instruction of procedure 1.2.2. In addition, the inspector noted that the proposed revision to maintenance procedure 3.M.3-7, "Inspection of Collector Rings and Armatures and Inspection & Replacement of Brushes on Rotating Motors, Generators & Fields," following the June 1 RPT did not address reactor power to flow prerequisites to initiate replacement of recirculation pump MG set brushes. Existing work controls did not effectively address the operational policies of procedure 1.2.2. These concerns were discussed with operations and maintenance section management.

Following a detailed review, the licensee initiated several corrective actions. Procedure 3.M.3-7 was revised to incorporate reactor power to flow prerequisites. Procedures 1.5.3 "Maintenance Requests" and 3.M.1-34 "Generic Troubleshooting Procedure," and the work request tag format were revised to specifically require the nuclear watch engineer or nuclear operations supervisor to assess the applicability of procedure 1.2.2 to the specified activity, prior to granting permission to commence work. The engineering department initiated a technical evaluation to better define the operating region from which the reactor would be susceptible to oscillations following an RPT. The evaluation concluded the previously specified region of "all area above the 80 percent rod line" was overly restrictive, which could result in unnecessary control rod motion. When approved, the licensee intends to incorporate the verified region of concern into procedure 1.2.2. In the interim, operations standing orders 92-09 "Procedure 1.2.2, Administrative OPS Requirements;" and 92-10 "Maintenance with Potential to Trip Recirc MG Set" were issued to provide guidance to the operators pending approved revision to the procedures previously discussed. Training was conducted for each operating crew. The inspector discussed the standing orders with several operators and determined that the training had been effective. Licensee corrective action was thorough and demonstrated appropriate safety perspectives. Inspector concerns regarding procedure 1.2.2 were properly addressed.

2.3 Condenser Bay Tour

The condenser bay area is periodically inspected visually for evidence of steam leaks, water leakage, or improper equipment performance. In addition to daily remote visual inspection via closed circuit television cameras, a weekly operator tour is conducted to provide better visual access. The licensee recently initiated utilization of a remote controlled robot with high powered camera and audio capability to perform the weekly condenser bay tours. The tour operator controls the robot and monitors the video/audio output from a remote location with a lower background radiation level. The operations staff intends to conduct operator condenser bay tours at a reduced frequency of approximately once per month. Implementation of ro... tours resulted in a noted reduction in personnel radiation exposure.

The inspector observed performance of a condenser bay tour conducted by the robot. Training of tour operators was conducted to develop proficiency at maneuvering the robot and associated camera features. The lighting, magnification, and clarity of the video output was outstanding.

Operators performed a detailed tour of the condenser bay which was not adversely impacted by time constraints of personnel exposure concerns. Introduction of robotic tours of high radiation areas is a positive initiative.

2.4 Reactor Water Cleanup System Leakage

On June 5, 1992, the licensee identified a small coolant leak dripping from an end plate bolt hole on the shell of regenerative heat exchanger E-203B in the reactor water cleanup (RWCU) system. Problem Report 92.0305 was issued and the steam leak was initially quantified by dripping condensation on the order of five drops per minute (dpm). Based upon similar leaks in this heat exchanger and a series RWCU heat exchanger, E-208A, in the 1989-1990 time fram., the current leak was believed to be attributed to a weld defect or crack in a stainless steel diaphragm serving as the heat exchanger pressure boundary, but non-load carrying member.

The RWCU system heat exchangers are nonsafety-related components which are isolated from the reactor coolant system (RCS) by a primary containment isolation system (PCIS) Group VI signal, initiated by high RWCU system flow or high area temperatures, among other safety signals. Thus, plant design and license conditions allow continued operations with the RWCU system isolated as long as reactor coolant system chemistry remains within operational limits.

Any safety concer. regarding an excessive RWCU leak is accounted for by the PCIS Group VI valve isolations which are designed to prevent any significant RCS leakage.

The current leakage of heat exchanger E-208B, which reached a steady state value of approximately 100 dpm, is well below any flow or temperature limitation controlling the Group VI PCIS system isolation. Operations personnel check this leakage each shift utilizing a remote camera positioned to directly view the leak in the RWCU heat exchanger room. The inspector accompanied an operations section manager on a plant tour to check the heat exchanger condition, via the remote camera capability, and confirmed leakage on the order of 100 dpm.

The licensee is planning to conduct repair activities on RWCU heat exchanger E-208B during any sustained plant shutdown conditions which would be conducive to such maintenance work in a high radiation environment. At the latest, such repairs will be effected during the mid cycle outage (MCO #9) in October-November, 1992. Until that time, the inspectors have verified th t continued safe operation of the facility is not adversely affected by the existing RWCU system leakage conditions. As appropriate, the inspectors will continue to monitor this leakage and will follow the planned repair activities with respect to the governing code provisions.

The inspector has no questions regarding the current status of this problem or the ongoing licensee monitoring activities.

2.5 Recirculation Pump Trip

On July 26 at 9:23 pm, the "A" recirculation pump spuriously tripped from steady state conditions while operating at 98% speed in manual control. The reactor quickly stabilized at approximately 60% of rated power as designed and single loop operations were conducted in accordance with procedure 2.4.17, "Recirculation Pump Trip." Control room operators inserted control rods as appropriate to ensure proper reactor core power to flow relationship was maintained below the 80% power rod pattern line.

Licensee electrical laboratory investigation determined that approximately ten seconds before the trip, the pump began to experience diverging minor speed oscillations and corresponding voltage oscillations. A decreasing or undervoltage condition was sensed by the motor generator set field breaker as a loss of field and the breaker tripped and the lockout breaker actuated as designed. Previously, the licensee has experienced "A" recirculation pump trips at high speed control settings. Voltage regulator tuning had improved high pump speed stability.

Upon verification that the pump tripped due to minor speed oscillations, the operators reduced reactor power to approximately 40%, established recirculation pump restart prerequisites and at 2:23 am on July 27, restarted the "A" recirculation pump. At 12:10 pm on July 27, the reactor returned to 100% of rated power. The operations section applied a caution tag to the "A" recirculation pump speed controller to limit speed to less than 96%. A similar caution is also applied to the "B" recirculation pump. The inspector had no further questions regarding licensee response to this event.

3.0 RADIOLOGICAL CONTROLS (71707)

3.1 Radiation Protection - Sludge Liner Movement

On July 15, 1992, the inspector observed portions of the movement of a radioactive sludge liner from shielding in the radwaste trucklock to a shipping cask on a flatbed trailer. The inspector checked the radiological control area (RCA) boundaries and postings for this operation, reviewed the radiation work permit (RWP 92-512), and interviewed several radiological protection (RP) technicians and supervisory personnel regarding the work controls. The shipping cask, identified as a Department of Transportation, DOT 7A - Type A container, was observed, after the cover installation, to be wiped clean with smear samples taken to confirm the acceptability of the outside surfaces. Smear samples of other material (e.g., the trailer tires) were noted to be obtained by RP technicians prior to transport of the shipping cask outside the radiologically controlled trucklock area.

The inspector also reviewed the Radiological/ALARA work plan for this activity, confirming with the radiological operations supervisors who authored the plan that this was the first sludge liner movement where the transfer of material occurring entirely within the radwaste trucklock. Previous sludge line transfers involved the potential for increased dose rates outside the trucklock and required an expanded RCA, including evacuation of a portion of the Administration Building exposed to the radioactive material shine. Thus, from an ALARA standpoint, the sludge liner movement that was witnessed represents an improved operation and a well planned activity.

Additionally, the inspector verified that QA coverage of this relatively infrequent (i.e., approximately once per year) radwaste activity was provided and that a pre-job meeting and worker briefing had been conducted. Plant management awareness and interest in not only the safe conduct of the operation, but also the personnel dose accumulations for ALARA and lessons-learned purposes was noted by the inspector during attendance at routine plant manager meetings. The inspector also noted that a contractor radwaste shipment truck driver, who was witnessing the sludge liner movement, was appropriately escorted by a member of the security guard force inside the protected area. The inspector confirmed that the flatbed trailer and shipping cask would receive an additional radiation survey, away from the RCA, prior to transport off-site.

The inspector had no questions regarding the observed sludge liner movement. Both radwaste and RP activities were well controlled and the supervisory and worker personnel were knowledgeable of the job requirements and established RCA criteria. The entire operation was effectively and efficiently conducted and no unresolved safety issues were identified by the inspector.

3.2 Dosimetry Issuance Program Revisions

The licensee recently initiated changes to the dosimetry issuance program for visitors who are not radiation workers and certain employees. Effective July 1, Level I general employee training (GET) employees and visitors are considered to be unmonitored workers and are no longer issued thermoluminescent dosimeters (TLDs) upon processing onsite. These individuals typically receive minimal exposure and historical records search indicates whole body exposures well below that which are required by 10 CFR 20 to be monitored.

In order to ensure compliance with regulations that require monitoring of people likely to exceed a whole body dose of 312 mrem per calendar quarter, unmonitored persons are required to wear self-indicating dosimeters (SIDs) in radiological control areas (RCA) and are limited to an exposure dose control limit of 25 mrem per calendar quarter as measured by SID. Further, the licensee objective is to ensure unmonitored personnel receive no more than 100 mrem per calendar quarter when not wearing dosimetry. The licensee will associate dose received by unmonitored personnel when not wearing dosimetry by; comparing Level II GET personnel TLD doses to SID doses; by reviewing restricted area surveys outside the RCA; and by reviewing doses measured by fixed position TLDs located in the restricted areas outside the RCA.

The inspector discussed the dosimetry issuance changes with the radiological section manager and independently reviewed affected radiological controls procedures and determined appropriate revisions had been completed. The inspector had no concerns regarding this subject.

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

4.1 Repair of Reactor Core Isolation Cooling Leak Detection Terminal Block

On June 16, 1992, Instrumentation and Control (I&C) technicians discovered a damaged (stripped) terminal contact point during performance of procedure 8.M.2-2.6.3, "RCIC Steam Line High Temperature." The damaged terminal point caused the RCIC valve station high temperature signal to the RCIC system isolation logic to become inoperable. Technicians halted the surveillance and informed the nuclear operation supervisor (NOS) of the damaged terminal point and disabled high temperature signal. The RCIC system was declared inoperable at 7:55 pm and the NRC was notified promptly in accordance with 10 CFR 50.72.

The NOS immediately initiated a priority one maintenance request (MR #19201894) to replace the damaged terminal point with one of the two installed spare terminal points on the terminal block. The location of the damaged terminal point was within the traverse incore probe room which was controlled as a locked high radiation area. The inspector discussed the intended repair work plan and radiological protection (RP) considerations with the electrical supervisor and found him to be highly knowledgeable. An RP specialist provided an excellent radiological briefing to the NOS and personnel who would be performing the repair. The work prebriefing was detailed and demonstrated careful planning. The quality control (QC) representative was alert in identifying that the torque device initially selected for the repair was of insufficient range. A replacement torque device of proper range and calibration was promptly obtained. Repairs were accomplished in accordance with procedure 3.M.1-34, "Generic Troubleshooting & Maintenance Procedure." Access to the locked high radiation area was properly controlled throughout the evolution and personnel radiation exposure was minimized. Surveillance 8.M.2-2.6.3 was successfully completed following repair of the terminal contact point. The RCIC system was declared operable at 2:00 am on June 19, 1992. Excellent coordination between operations, electrical, I&C, RP, and QC was effective in minimizing the duration of RCIC system inoperability.

During the performance of MR #192001894, electricians noted that the configuration of the terminal contact points within electrical junction box J599, was not consistent with reference drawing E224 (sheet 144). Maintenance request #192001895 was initiated to restore the correct terminal contact point configuration. This MR was planned and implemented in an effective manner which did not involve removing RCIC from service.

4.2 Non-Code Repair to Salt Service Water System Piping

On November 14, 1991, a test engineer found a wooden plug installed in a 3/4 inch pipe flange on an ASME Class 3 section of the "B" salt service water (SSW) system loop discharge piping. This piping had previously functioned as a sodium hypochlorite sample tap, but was abandoned in place with the implementation of Plant Design Change 86-52B-146. The wooden plug repair did not meet code standards for ASME Class 3 components as required by 10 CFR 55(a). In addition, contrary to NRC Generic Letter (GL) 90-05 "Guidance for Performing Temporary

Non-Code Repair of ASME Code Class 1,2, and 3 Piping," this non-code repair had been implemented without requesting written relief from the NRC. Use of a wooden plug was an improper application to repair the SSW system. The licensee could not locate any maintenance documentation to indicate that the wooden plug repair had been authorized or when it was performed.

Upon finding the wooden plug installed, the test engineer initiated the corrective action process by writing a Work Request Tag (WRT). Contrary to procedure 1.5.3 "Maintenance Requests (MR)," the WRT was not reviewed and processed into an MR in a timely manner. Despite being categorized as a priority 2 MR (#19104873), maintenance planners incorrectly scheduled the MR to be completed during the next refueling outage. Quality Control (QC) personnel reviewed the MR package on June 22, 1992 and determined that the existing condition (wooden plug installed in ASME Code 3 piping) was a non-code repair. The reviewer initiated a problem report (PR 92.0323) to address the non-conforming condition.

The inspector noted that a separate MR (#19101893) had been previously initiated on March 13, 1991, which identified a leaking component a short distance below the wooden plug location. A job planning walkdown for this MR found the problem to actually be the wooden plug installed above the initially reported problem area. A maintenance planner incorrectly assumed that the affected piping would be replaced during the performance of an upcoming plant design change (FRN 85-28-10). The planner then closed MR 19101893, referencing the existing open MR (#19101677) which would be completed during installation of FRN 85-28-10. The inspector determined that the process by which an MR can be closed and transferred to another open MR was not properly implemented. In addition, the maintenance staff did not recognize that the affected piping was ASME Class 3 and did not disposition the MR in a time period commensurate with the importance of the repair.

The licensee corrective action processes were not properly implemented in response to this issue. Initially, at some time prior to November 14, 1991, an unauthorized repair activity was initiated to a safety-related system that was identified neither by proper corrective action process documentation nor by report of the deficiency to appropriate levels of management. Additionally, because corrective action process controls were not enacted, the unauthorized repair was not in conformance with applicable ASME Class 3 component code standards as required by 10 CFR 50.55a or with the recommendations of NRC GL 90-05. Further, after identification and documentation of the unauthorized repair on November 14, 1992, corrective actions were not implemented for approximately seven months. During this period, several opportunities existed to address and correct the deficient condition. However, due to the lack of aggressive followup by involved licensee personnel the ASME code implications were not identified until QC review of the MR on June 22, 1992. Failure of the corrective action processes to identify, document, and properly correct the SSW system deficiency in a timely manner is a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action" (VIO 50-293/92-14-01).

The unauthorized non-code installation of the wooden plug to the SSW system was identified by the licensee, was of minimal safety significance, and did not preclude system operation. However, because licensee corrective action processes failed to ensure prompt correction of the deficiency once identified and documented, a citation is being issued notwithstanding the guidance in Section G of the Enforcement Policy regarding exercise of discretion.

Subsequent, root cause analysis and recommended corrective actions have been thorough. Immediate corrective action included an approved code repair which was completed on June 25, 1992 and a complete SSW system walkdown which verified that there were no additional non-code repairs installed in the system. Additional long term corrective actions included training of various members of the licensee organization and contractors on procedures 1.5.3 and 1.5.9 "Temporary Modifications" and the recommendations of NRC GL 90-05. The maintenance section manager initiated reviews of several procedures to clarify, as appropriate, several aspects of the work control process and individual responsibilities. Initial corrective actions have been appropriate.

4.3 Routine Surveillances

Procedure 8.5.5.9, "Reactor Core Isolation Cooling (RCIC) Simulated Automatic Actuation, Flow Rate and Cold Quickstart Test," was performed on July 10, 1992. This once per operating cycle procedure was performed to verify cold start capability and system operability following an automatic actuation signal. The inspector monitored this surveillance from the control room. The preevolution briefing and work assignments were clear. As documented in NRC Inspection Report 50-293/92-08, some operators had previously experienced difficulty positioning the full flow test valve (MOV 1301-53) to establish the desired flow rate. The inspector noted that the operator had no difficulty establishing the specified system flow rate at which performance data was collected. Operators were knowledgeable regarding the procedure and annunciators which actuated during the surveillance. Second verification of valve positions during system lineup restoration was noted to be strong. The surveillance was completed with satisfactory results.

5.0 EMERGENCY PREPAREDNESS (71707, 40500)

During the evening of June 24, a contract training instructor exhibited symptoms of cardiac arrest while in the control room. Station emergency medical technicians responded to the scene and the nuclear watch engineer directed that assistance of the Town of Plymouth ambulance be requested. Procedure 5.5.3, "Medical Emergency Response Procedure," was properly implemented. The ambulance responded to the station and Town of Plymouth EMTs administered to the stricken instructor and when stabilized transported the individual to the Jordan hospital for continuing medical attention.

The licensee responded in a prompt and well controlled manner to the medical emergency. Communications with the Town of Plymouth were properly implemented from the control room. The security force quickly mobilized to expedite ambulance entry and exit from the protected area. The inspector had no concerns regarding this event.

6.0 SECURITY (71707)

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify that controls were in accordance with the security plan and approved procedures. This review included the following security measures; security force staffing, vital and protected area barrier integrity, maintenance of isolation zones, behavioral observation, and implementation of access control including access authorization and badge issue, searches of personnel, packages and vehicles, and escorting of visitors.

On June 24 during deep backshift inspection of the facility, the inspector conducted extended observation of the secondary alarm station (SAS) performance. The SAS security force officer was fully aware of perimeter intrusion detection system and surveillance system status. The officer verified that acknowledgement and response to alarms were promptly and properly conducted. Additionally, the transfer of alarm station command from the central alarm station (CAS) to the SAS during CAS turnover and subsequent return of command to CAS was completed in a well controlled manner with clear and formal communications noted. The inspector had no questions regarding this inspection activity.

7.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (92701, 40500)

7.1 Organizational Restructuring

On June 29, the licensee initiated the first phase of a planned three phase organizational restructuring initiative. The initiative is in response to a utility-wide employee survey and an external manpower study. The changes are intended to strengthen organizational support of facility operations and to improve organizational communication and efficiency.

Two significant phase I changes included reassignment of the Emergency Preparedness Department to report to the Vice President (VP) Technical (formally VP Engineering) vice the Senior Vice President Nuclear and creation of the Director of Nuclear Engineering position that reports to the VP Technical with responsibility for management of the Nuclear Engineering Department, the Material and Component Engineering Section, and the Projects and Construction Section.

Phase II of the reorganization is scheduled to be implemented during the first quarter of 1993, and is projected to be in place through the next refueling outage. Phase III, the final phase, is scheduled to be implemented in July of 1993 following completion of the refueling outage.

Inspector review of administrative sections of the Technical Specifications concluded that no amendments were necessitated by Phase I organizational changes.

7.2 Licensee Report Review - Special Report - Inoperable Diesel Fire Pump

On June 30, in accordance with Technical Specification 3.12.B.a, the licensee submitted a special report to the NRC documenting that the diesel driven fire pump of the fire suppression system was inoperable for a period of more than seven days. The pump had been removed from service on June 1, for various maintenance inspections and to replace the overspeed trip switch. During the planned pump outage, a leak was identified in the fuel oil manifold that necessitated manifold replacement. The licensee did not have a manifold in stores and therefore was required to procure a replacement from the manufacturer. The new manifold was received and installed June 11. Post maintenance testing problems with the overspeed trip switch function delayed return to service of the pump until June 16. The special report fulfilled Technical Specification requirements.

8.0 ENGINEERING AND TECHNICAL SUPPORT (71707, 37828)

8.1 Salt Service Water Piping Replacement

As documented in inspection report 50-293/92-08, the licensee commenced the salt service water (SSW) piping replacement project in May 1992 with the initiation of construction activities governed by plant design change (PDC) 91-10A. Additional design changes, PDCs 91-10B, 10C and 10D have been issued to establish the criteria and control sequential work involving concrete vault construction, penetrations into the existing SSW intake structure and auxiliary building bays, and installation of the major portion of the new titanium piping supply and discharge headers. During this inspection period, PDC 91-10E was issued and is scheduled to be implemented during the next refueling outage (RFO #9) to provide the final tie-in of the new buried titanium piping into the existing SSW loops. The inspector reviewed PDC 91-10E and the associated Safety Evaluation, SE-2684, verifying the appropriate consideration of ASME Section XI requirements, train related work sequencing, and the use of as-built measurements to ensure precise spool piece fabrication and alignment for seismic integrity.

The inspector also reviewed nonconformance reports, NCR Nos. 92-27, 92-28, 92-29 and 92-30, regarding material either installed or affected by ongoing PDC 91-10A construction activities. Engineering evaluation of the nonconforming conditions for both structural adequacy and design impact was noted. The inspector confirmed that in the case of NCR 92-29 concerning wire rope capacity testing, additional load tests were conducted prior to the cable being subjected to any safety-related loading applications in the field. The inspector also discussed with nuclear engineering division (NED) personnel the status of meetings with the titanium piping supplier to resolve certain questionable test coupon tensile testing results.

The inspector observed construction activities related to PDC 91-10A stage 1 excavation and steel support framework installation, south of the SSW intake structure. The inspector witnessed structural beam fillet welding, gusset/stiffener plant installation and wire rope positioning in preparation for structural support slinging of the Appendix R electrical duct banks. Shielded metal arc weld (SMAW) rod material and the final condition of specific fillet welds were inspected and the wire rope clip configuration and spacing details for one cable sling were checked. The inspector verified conformance with standard AWS D1.1 structural welding and wire rope rigging practices.

The overall SSW piping replacement project schedule, procurement activities and phased controls were discussed with cognizant NED personnel. Specific questions on observed PDC 91-10A construction activities were discussed by the inspector with construction engineering, field QA, and contractor work force personnel. No unresolved safety issues or concerns with the control and progress of ongoing construction were identified. At the conclusion of this inspection period, further stages of excavation and support work, governed by PDC 91-10A, were in progress. The inspectors will continue to conduct followup evaluations of the SSW piping replacement project, as appropriate.

8.2 Review of Engineering Topics at NED

On July 2, 1992 the inspector met with cognizant NED personnel in Braintree, Massachusetts to discuss engineering activities related to the following subjects:

-- Salt Service Water (SSW) pump improvements (reference: Bid Specification M-8B)

-- SSW piping replacement activities and schedular plans (see section 8.1 of this report for additional inspection details)

-- Engineering evaluation of LER 92-01 and Code Case N411 compliance, with a representative case analysis of a scram discharge volume design change (reference: unresolved item 50-293/92-04-01)

-- Engineering evaluation of the need (heat load efficiency, not safety-related considerations) for dredging the SSW intake canal

-- Status of Individual Plant Examination (IPE) and IPE of External Events (IPEEE) activities (reference: responses to Generic Letter 88-20 with supplements)

-- Potential midcycle outage no. 10 plant design changes and NED initiatives

The inspector evaluated the above discussion topics with regard to the current plant status (e.g., SSW piping) and licensee commitments. The requirement to address certain technical issues (e.g., UNR 92-04-01) prior to completion of RFO #9 activities was well understood. The inspector had no questions regarding either the priority or the approach to technical issues currently within the preview of the licensee NED. The inspection review resulted in a valuable exchange of information regarding future engineering activities affecting the PNPS. No additional unresolved safety issues were identified.

9.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (30702)

9.1 Routine Meetings

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

9.2 Management Meetings

On July 1, a licensee and NRC management meeting was conducted in Rockville, MD. at the NRC Headquarters at One White Flint North to discuss several near term license amendment requests.

On July 21, a licensee and NRC management meeting was conducted in King of Prussia, PA at the NRC Region I office to discuss licensee security section activities and initiatives.

9.3 Other NRC Activities

On June 22-26, an NRC confirmatory measures inspection was conducted to evaluate licensee proficiency at obtaining and analyzing radioactive liquid and gaseous samples. Inspection results will be documented in Inspection Report 50-293/92-13.