U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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Report No.:	50-293/91-12
Licensee:	Boston Edison Company 800 Boylston Street Boston, Massachusetts 02199
Facility:	Pilgrim Nuclear Power Station
Location:	Plymouth, Massachusetts
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Inspectors:	J. Macdonald, Senior Resident Inspector A. Cerne, Resident Inspector P. Sena, Reactor Engineer, DRP W. Baunack, Senior Reactor Engineer, DRS F. Bower, Reactor Engineer, DRS

Approved by: John F Rogge 7/30/91
John Rogge, 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.

Unresolved Item: An unresolved item was issued to assess the effectiveness of licensee critique and corrective actions to radiological protection program weaknesses identified in review of several personnel contamination events (UNR 50-293/91-12-01, Section 3.1).

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EXECUTIVE SUMMARY

Pilgrim Inspection Report 50-293/91-12

Plant Operations: Control room operators continued to maintain excellent command and control of available station power supplies and plant system configurations throughout the diverse outage testing and maintenance evolutions. Notwithstanding, nuclear watch engineer authorization of conflicting plant activities caused an inadvertent engineered safety feature system actuation.

Licensee investigation of the dropped fuel bundle event was deliberate and comprehensive. Potential failure mechanisms were identified and evaluated. Suspect components were replaced. Final analysis indicated potential human error contribution to the event occurrence. Human factors enhancements were installed on the refuel bridge control system to address this potential.

Radiological Controls: Three personnel contaminations occurred during work activities performed on the refueling bridge during the period June 27-28, 1991. Evaluation and decontamination of these individuals was performed properly. However, several inspector concerns regarding ineffective implementation of the existing radiological controls program have evolved in relation to the contamination incidents.

Maintenance and Surveillance: Plant and maintenance section management continue to provide timely resolution to areas of concern identified during the NRC Maintenance Team Inspection. Continued attention is necessary to ensure effective management plant tours during outage conditions as well as during general system engineering system walkdowns. Additionally, although improved, further material condition enhancements remain to be realized in the screenhouse and condenser bay.

Emergency Preparedness: Procedural revision and new procedure issuance have appropriately addressed previous communications difficulties encountered during events which required assistance of offsite emergency response organizations.

Security: The security section continued to effectively implement the security plan. Vehicle movement within the protected area was well controlled.

Safety Assessment and Quality Verification: Licensee event reports issued were timely and accurately described event sequences, safety consequences, and corrective actions. Station management displayed conservative safety perspective in support of resolution of salt service water (SSW) system inspection scope and spool piece replacements.

The multi-disciplinary analysis team (MDAT) investigation of the degrappled fuel bundle was noteworthy. Onsite review committee (ORC) review of MDAT investigation findings was generally adequate. However, ORC did not deeply probe into the appropriateness of continued bundle movement after unsuccessful insertion attempts in the reactor core. Continued inspector review of this question identified varying management expectations in this area. Subsequently station management positively and clearly stated expectations with respect to fuel movement activities.

Engineering and Technical Support: The nuclear engineering department (NED) provided continuous support to the interpretation of SSW pipe inspections. Recommendations to station management for inspection corrective actions reflected a conservative safety orientation. Emergent outage issues, including standby liquid control (SLC) system squibb valve discrepancies, were effectively dispositioned in a timely manner. Additionally, continued attention to long standing complex NRC inspection findings facilitated issue closure.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

At the beginning of the report period, Pilgrim Nuclear Power Station (PNPS) was in Cold Shutdown, with the reactor core off-loaded, and Refueling Outage #8 activities ongoing. The "A" train of safeguards systems were removed from service for maintenance and surveillance testing. The "B" loop of the residual heat removal (RHR) system was in the augmented fuel pool cooling mode of operation. Additionally, detailed control room design review upgrades and modifications were in progress throughout the report period.

On June 9, 1991, the "A" loop of RHR was returned to service and "B" train systems were prepared to be removed from service for maintenance and testing. On June 19, 1991, the "A" loop of the core spray system and the 4160 V A-5 bus were returned to service and declared operable. Reactor core reload was commenced on June 21, 1991.

On June 26, 1991, a spent fuel bundle became ungrappled from the refueling bridge mast while being placed in a vacant fuel rack location in the spent fuel pool. The bundle came to rest in the appropriate vertical orientation at the selected rack location without radiological consequence (section 2.3). Following licensee investigation of the event and implementation of human factors enhancements to refueling bridge control systems, refueling operations were restarted on June 29, 1991. Reactor core reload was completed on July 2, 1991 at which time core verification was initiated and subsequently completed. On July 3, 1991, shutdown margin verification testing was completed satisfactorily.

On May 27, June 4, and June 7, 1991, the licensee notified the NRC Operations Center via the Emergency Notification System (ENS) of inadvertent engineered safety features system actuations during various surveillance testing evolutions (section 4.2). The notifications were completed in accordance with the requirements of 10 CFR 50.72 reporting criteria.

2.0 PLANT OPERATIONS (71707, 71710, 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 Building
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; May 28-31, June 4-6, 10-12, 17-20, and July 2-3. Deep backshift inspection was performed on May 27 from 9:30 am to 1:30 pm.

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

2.2 Review of Tagging Operations

The tagout log and tagging activities were inspected to verify plant equipment was controlled in accordance with the requirements of station procedure 1.4.5, "PNPS Tagging Procedure." During this inspection the following tagouts were reviewed with no discrepancies noted.

Tagout	Description
T91-10-55	"B" RHR pump shutdown cooling suction valve EQ maintenance
T91-6-20	Turbine Building Closed Cooling Water to feedwater pump "A"
T91-10-57	Suppression chamber fill
T91-61-23	Power supply for diesel generator "B" control and field flashing
T91-61-22	Power supply for "B" diesel generator DC emergency oil pump

Station procedure 8.A.25, "Periodic Review of Tags" provides instructions for the periodic audit of various protective and non-protective tags as required by PNPS 1.4.5. A review of completed audits of caution tags, master danger tags, and tagout sheets was performed on a sampling basis with no discrepancies noted.

2.3 Dropped Fuel Bundle

On June 26, 1991, at approximately 4:00 am, while positioning spent fuel bundle LJX705 into a vacant rack location in the spent fuel pool, the bundle became unlatched from the refuel bridge handling device. The bundle dropped freely approximately 11 feet into the appropriate fuel storage rack location. All unnecessary personnel evacuated the refuel floor. However, radiological surveys immediately verified no release of radioactivity or any increase in radiation levels. The licensee secured all further fuel movement operations and formed a Multi-Disciplinary Analysis Team (MDAT) to review the occurrence and identify the root cause.

Fuel bundles are transported between the spent fuel pool and reactor core via the refueling bridge. A telescoping mast and grapple assembly is mounted to the bridge. The grapple is comprised of three opposed "J" hooks which actuate on air pressure to latch the fuel bundle bail handle. The grapple is designed such that it cannot be opened while the weight of the fuel bundle is loaded on the "J" hooks, even with an open signal applied. The irradiated bundle was being returned to the spent fuel pool following unsuccessful attempts to place the fuel bundle in the reactor core. The fuel grapple mechanism had failed to unlatch the bundle in the core on three previous attempts. It was determined that interference between a mast mounted camera and the core shroud prevented the grapple from physically unlatching.

The MDAT thoroughly investigated all possible mechanical failures which could have caused or contributed to an inadvertent unlatching of the bundle. The inspection and testing conducted eliminated the mechanical failure mechanisms under consideration as the root cause, leaving personnel error as a possibility. However, as a precaution, the licensee replaced all mechanical components of the grapple and grapple actuator.

Although personnel error could not be confirmed as the root cause, the MDAT postulated that the grapple engage/release switch may have been left in the release position by the operator following the third unsuccessful attempt to ungrapple the bundle in the reactor core. With this switch in the release position, the fuel would not be released during transit back to the spent fuel pool due to the weight of the bundle on the "J" hooks. Additionally, the grapple position lights would indicate grapple engaged. As the fuel was being positioned in its spent fuel rack location, the MDAT again postulated that the grapple opened when the fuel engaged the storage rack (relieving a significant portion of the bundle weight), moving the bail handle up and out of the way of the "J" hooks, thus allowing the hooks to open with the bundle unloaded. The inspector concluded this scenario was plausible, given the results of the root cause analysis. The licensee subsequently installed pressure gages on the air supply lines to the grapple hook to provide positive visual indication of a grapple release or engage signal to the operators. The senior reactor operator (SRO) and reactor operator (RO) performing fuel movement are now required to verify grapple switch position, remote camera grapple position indication, and grapple actuating air signal following every grapple/ungrapple evolution. Additionally, procedural changes were implemented to preclude any possible future camera/shroud interference.

The inspector observed the MDAT presentation of their findings to the Operational Review Committee (ORC). Although no positive root cause could be identified, the MDAT investigation was regarded by the inspector to be extremely thorough and meticulous. The ORC review of the event was adequate, however, the issue of repetitive attempts to insert the fuel bundle in the reactor core was not addressed by the ORC. The inspector questioned the appropriateness of continued movement of bundle LJX705 following the inability to ungrapple the fuel in the reactor core and received conflicting answers from licensee management and licensed operators. The inspector concluded that increased communication of managerial expectations was less than adequate in this instance. This concern has been properly addressed by senior licensee management and is considered resolved.

Fuel handling evolutions were observed by the inspector prior to and following this event. All fuel movement operations were supervised by a licensed SRO stationed on the refuel bridge and conducted by a licensed RO. All fuel moves were conducted by procedure and were independently verified by a second observer. The SRO informed the main control room of any fuel bundle being grappled or ungrappled. The procedural changes initiated as a result of this event were observed to be properly implemented. Following resumption of fuel movement operations reactor core fuel reload was completed without incident. Spent bundle LJX705 was not returned to the core and was retired to storage in the spent fuel pool. The inspector had no further questions regarding this event.

2.4 Standby Liquid Control

During the conduct of procedure 8.4.6 for the standby liquid control (SLC) system in-circuit squib valve firing on June 12, 1991, an extraneous part (i.e., the rupture disc from a previous squib valve firing surveillance) was found within the valve body of SLC valve 1106A. This in-circuit firing was performed in accordance with Attachment 1 to the procedure for the manual initiation test of the SLC system, which test fires an already installed squib charge without injecting water into the reactor vessel. Engineering Service Request (ESR) 91-0404 and failure and malfunction report (F&MR) 91-242 were initiated to evaluate the component impact and operational aspects of this discovery. Additionally, reportability in accordance with 10 CFR 50.72 was evaluated.

The inspector observed a licensee critique of this situation on June 13, 1991. At that time, the surveillance test results from previous firings of both trains of squib valves (i.e., 1106A and B) were discussed. It was noted that valve 1106A had previously been fired in November 1988, with documented evidence that the rupture disc from that test firing had been removed from the valve body. Since the prior test firing in September 1987 did not have the same evidence of rupture disc removal after completion of the surveillance test and since the disc that was found exhibited markings consistent with having been fired upon more than once, the licensee suspected that this extraneous disc had remained in the body of valve 1106A for more than one cycle. Also, since acceptable flow criteria were achieved during the conduct of surveillance procedure 8.4.6 in 1988, the presence of this disc anomaly was determined to have not adversely affected the operability of the "A" train of the SLC system during the previous operating cycle.

The licensee critique evaluation group confirmed the fact that engineering analysis was in progress (i.e., ESR 91-0404) and determined that contact with the squib valve vendor and in-house procedural review was required to evaluate the installed valve condition for future operability, as well as the adequacy of existing test controls. The licensee also confirmed that this deficiency did not exist in the redundant "B" train SLC squib valve, 1106B.

The inspector witnessed the conduct of section 7.2 of procedure 8.4.6 on June 14, 1991 to check the squib valve firing system, SLC pump capacity, and flow to the reactor vessel. The inspector observed operator initiation of this test including isolation of the reactor water cleanup system and verified the acceptability of the resultant control room indications, to . The inspector also confirmed the adequacy of control room component and indication status markings, given the ongoing control board improvement activities; checked the test conditions and controls at the squib valve location to verify proper component and boundary tagging; witnessed licensee tagout verification activities after completion of this portion of the surveillance procedure; and interviewed operations personnel both in the control room and in the SLC pump/valve area in the reactor building to discuss overall test conduct and criteria. The inspector noted that an acceptable pump flow rate was achieved, passing the surveillance criteria for the "A" train SLC system in accordance with Technical Specification 4.4.A.2.c. Previously, the "B" train SLC system had been successfully tested during the 1990 midcycle outage.

After completion of the squib valve firing and SLC train "A" flow test, the valve body was inspected and the explosive valve trigger mechanism was replaced for valve 1106A. The inspector reviewed ESR response memorandum ERM 91-401, documenting the engineering justification for the acceptability of the valve body for 1106A, given that the extraneous rupture disc had remained in place for one or more cycles of operation. The ductility of the stainless steel material, as well as its impact strength, and consideration of the valve body wall thickness, which is approximately six times the minimum required wall thickness in the area where the rupture disc was found, all provide conservative margins of protection against a through wall leak developing as a result the identified rupture disc anomaly. Additionally, the inspector noted that the subject valve vendor (Conax Buffalo) had been contacted on this issue and had indicated that the multiple firings of a squib valve with a rupture disc remaining in the valve cavity should have no deleterious effect upon the structural integrity of the valve body.

The inspector reviewed portions of the Maintenance Request (MR 19100405) documenting successful completion of the exterior inspections of the valve body (1106A), as required by the disposition of ERM 91-401, with the results indicating that no obvious defects were detected. The remainder of surveillance testing of the SLC system (e.g., the nitrogen accumulator pressure test) was completed in accordance with procedure 8.4.6. The successful conduct of this procedure and the restoration of both trains of the SLC system to service supported an acceptable operability determination prior to the commencement of refueling activities.

The inspector evaluated the overall conduct of the SLC surveillance activity, to include discovery, critique, engineering assessment, and corrective actions taken relative to the anomalous rupture disc condition identified in valve 1106A. The licensee resolution of this issue was methodical and controlled. Vendor guidance was utilized in the engineering evaluation and additional inspection was conducted. The inspector identified no unresolved safety concerns and had no questions relative to the return of the SLC system to operable status.

3.0 RADIOLOGICAL CONTROLS (71707)

3.1 Personnel Contaminations During Work on the Refueling Bridge

Three personnel contaminations occurred during work activities performed on the refueling bridge on June 27 and 28, 1991. One contamination incident involved a hot particle and another resulted in a positive uptake of radioactive material. The inspector observed a critique of these incidents and reviewed a draft of the critique report.

Subsequent to the critique, the inspector discussed three areas of concern with the licensee: (1) methods for using whole body friskers upon exiting contaminated areas; (2) control of face shields; and (3) adequacy and effectiveness of hot particle control zone (HPCZ) training implementation.

The licensee critique identified that some personnel don personal clothing before using the whole body friskers and a number of personnel expressed confusion concerning the requirements for using whole body friskers upon exiting a contaminated area. Discussions with the licensee found that the preferred method is to use the whole body frisker prior to donning personal clothes; however, station radiological protection procedures do not specify frisking requirements to this level of detail. The licensee has taken action to inform workers and managers of the contaminated area frisking requirements and further corrective actions are under review.

During the critique, workers indicated that the methods of controlling face shields in the refueling floor contaminated area may have contributed to facial contaminations. It was revealed that workers piled the used face shields at the contaminated area entrance/exit point prior to exiting the area. Instead of using clean face shields, workers entering the contaminated area may have reused these potentially contaminated face shields after they had been wiped down by HP technicians. Discussions with the licensee found that the control of face shields is not proceduralized. Although not proceduralized, the licensee has changed their system for handling face shields. Only clean face shields are to be used to enter a contaminated area and face shields are to be placed in receptacles at the contaminated area exit point for decontamination prior to reuse. The licensee has taken action to inform personnel of this new system and further corrective actions are under review.

During the licensee critique one of the workers contaminated at the refueling floor area stated that he had not received HPCZ training and was not aware of the requirement for this training. This worker was the individual who received a positive uptake of radioactive material. The licensee completed a bioassay of this individual and determined that the total uptake was below the regulatory limits established in 10 CFR 20. Station procedure 6.1-016, "Hot Particle Contamination Control Program," requires workers to be trained prior to working in a HPCZ. The licensee indicated that the General Employee Training (GET) satisfies this requirement. The contaminated worker had successfully completed GET.

Procedure 6.1-016 also provides an attachment, "Checklist for Considerations of Radiological Controls when Planning for Work With Hot Particles," which is required to be prepared and attached to HPCZ radiation work permits (RWP). Radiological controls personnel evaluate the work to be performed, select appropriate precautions from the checklist, and mark the applicable precautions to be implemented for the subject RWP. Workers received the small source briefing required by the completed procedure 6.1-016 checklist attached to the refueling floor RWP. Several additional radiological control considerations available on the RWP checklist attachment (i.e. providing specialized training in protective clothing removal, use of friskers, and contamination control techniques) were not marked as applicable. When completing this checklist for the refueling bridge work activities, the licensee determined that these additional types of training were not needed. The inspector finds that requiring this additional training would have been prudent and may have precluded the face shield and frisking contamination incidents described above.

The licensee also indicated that "enhanced" HPCZ training developed for the current refueling outage was intended to be provided in addition to the training required by procedure 6.1-016. The inspector noted that this training program was not formalized or proceduralized. The licensee intended to use an access list to limit the workers allowed to enter an HPCZ to those having received the "enhanced" HPCZ training. However, discussions with the licensee revealed that 15 workers who were not documented to have had the "enhanced" training were inadvertently added to the access list. One of these workers was contaminated while working within the refueling floor HPCZ. The licensee has taken corrective action to include only those workers documented as having received the "enhanced" HPCZ training on the access list, but no action has been taken to formalize or proceduralize this program. The licensee is reviewing further corrective actions

The three concerns described above are specific examples of ineffective implementation of the radiological controls program. Licensee resolution of these concerns is in progress. This issue is considered unresolved pending issuance of the final critique report, identification, and evaluation of the licensees final corrective actions (UNR 50-293/91-12-01).

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

The inspector observed portions of surveillances to verify proper calibration of test instrumentation, use of approved procedures, performance of work by qualified personnel, conformance to limiting conditions for operation (LCO), and correct system restoration following testing. The following surveillances were observed:

-- Procedure 8.9.16.1, Manually Start and Load Blackout Diesel via the Shutdown Transformer, Revision 1, effective April 21, 1990, observed July 2, 1991.

The surveillances on the diesel generator were performed well by plant personnel. During testing, the inspector noted good management oversite of activities at the diesel generator. No unacceptable conditions were identified.

-- Procedure 9.16, Shutdown Margin Check, Revision 13, effective June 27, 1991, observed July 3, 1991.

The shutdown margin check, conducted to meet the requirements of Technical Specification 3.3.A.1, ensures that the core can be made subcritical in its maximum reactivity condition with the most reactive control rod fully withdrawn and all other rods fully inserted. A pre-evolution brief was conducted and all precautions and limitations were discussed. Activities in the control room during the surveillance were professional, with good communications exhibited between workers. The inspector noted good management safety perspective as evident by the management oversite of control rod manipulations. Sufficient shutdown margin was demonstrated and no deficiencies were identified.

4.1 Maintenance Team Insp... (MTI) Followup (92701)

An MTI was conducted at the Pilgrim Nuclear Power Station from November 5-16, 1990 and December 10-20, 1990. During that inspection, two weaknesses were identified that included the overall planning and supervision of maintenance activities and the procedure review process. The licensee provided a written response to these weaknesses by letter dated April 30, 1991. During this inspection the actions taken by the licensee to correct the weaknesses were examined.

Also, during the MTI, issues were noted which were not categorized as weaknesses. These issues included deficiencies noted by the team and other areas in which the licensee was in the process of making improvements which the team considered in making their findings. Progress made by the licensee in these areas was also reviewed during this inspection, along with the observation of maintenance activities in progress.

4.1.1 Licensee Action on MTI Identified Weakness Involving the Planning and Supervision of Maintenance Activities

This weakness dealt with the overall planning and supervision of maintenance activities. Several licensee initiatives intended to improve and strengthen this area were inspected and are described below.

-- First line supervisor responsibilities relating to the assurance that job packages are fully ready to work have been reduced in order for them to provide increased oversight of ongoing maintenance activities. No procedure changes were necessary to achieve this since the responsibility for job ready packages had always been a planning function. Based on discussions with licensee personnel, a refinement of procedural responsibilities has resulted in improvement in this area. However, need for further improvement during outages has been recognized by the licensee. One procedure change which had an affect on the preparation of job ready packages was to clarify responsibilities for the maintenance of consumable open stock material.

-- Training of maintenance supervisors has been enhanced to increase their supervisory knowledge and skills. The core training requirements for maintenance supervisors were verified to have been included in the Maintenance Section Manual. Also, prior to the outage a one day INPO Observation Training Course was provided to maintenance supervisors to improve their oversight of job activities. The licensee is anticipating further improvements in the conduct of maintenance with the initiation of twice weekly training sessions.

-- The production work force has been restructured into teams, each assigned to a specific first line supervisor. This has been in effect since the first of the year. Due to the scope of activities and duration of an outage condition, the team concept has been found not to function well during the current outage. However, teams will be maintained following the completion of this outage with further improvement in work quality and productivity using this concept expected.

-- The efficiency of the work control process has been increased by a change to the Maintenance Request procedure which added the Work Request Tag (WRT). One advantage of the WRT is that, with certain specified controls, minor work items on non-safety components not affecting plant operations may be completed on the WRT. This eliminates the need for processing a maintenance request for these minor items and makes better use of available resources.

-- Another initiative to improve the work control process has been the establishment of a program to reduce rework and recurring maintenance. This program is implemented through procedure 1.5.3.2, "Rework Maintenance Evaluation," which provides for root cause analysis and corrective action.

-- To further improve, the Maintenance Quality Improvement Program has been initiated. The procedure for the implementation of this program has been added to the maintenance section manual and is focused on the actual observation of work activities. This program is a joint effort by Maintenance, Planning and Outage, and the Quality Assurance Departments to identify and resolve concerns so as to improve the quality of maintenance work performed.

Overall the licensee has taken some steps intended to improve the planning and supervision of maintenance. Additional measures have been established which monitor the control of maintenance to identify areas where further improvements can be made. This weakness is considered to be adequately addressed.

4.1.2 Licensee Action on MTI Identified Weakness Dealing with the Procedure Review Process

This weakness concerned the procedure review process that failed to identify certain deficiencies that still existed when the lubrication sampling and changing procedure was revised. The licensee has taken the following steps to address this issue:

-- Administrative procedure 1.3.4, "Procedures," was verified to have been revised to require that technical review be performed on all new procedures. Also, procedure 1.3.4-4, "Procedure Technical Review and Validation," was revised to add section 5.5, Technical Reviews. This procedure provides acceptance criteria to be used during performance of technical reviews. Included in these criteria are assurances that quantitative and qualitative acceptance criteria are included in the procedure.

-- Maintenance procedure 3.M.4-17.4, "Lubrication Sampling and Change Procedure," was revised to add specific acceptance criteria and to provide instructions for preventive maintenance coordinators responsibilities relative to the evaluation and notification requirements regarding lube oil analysis reports.

The licensee has taken adequate measures to address this weakness.

4.1.3 Licensee Actions on Other Issues Noted in the MTI

Issue #1: The team noted that walkdown efforts lacked a formal procedure to integrate the plant walkdowns performed by various site groups. To resolve this matter the licensee issued procedure 1.3.103, "Management Plant Tours and Inspections." This procedure provides guidelines to be followed and delineates responsibilities for the implementation of the plant material condition inspection program. The goal of this program is that an inspection of the entire plant be completed on a bimonthly basis. Inspection teams have, as a minimum, representatives from; operations, technical support, maintenance, radiological protection, and station services. The program has provisions for the documentation and correction of deficiencies identified.

During the MTI the team also noted that the screenhouse and to a lessor degree the auxiliary bay area, in contrast to the rest of the plant appeared deteriorated. The licensee indicated that the implementation of the management tours inspection procedure would prevent similar conditions from developing elsewhere and would serve to maintain the entire plant at the standards that management expects.

During this inspection it was observed that several job sites were being maintained in an untidy condition and that certain radiological controls could be improved. The inspector noted the management plant tours inspection program appeared to be directed more toward the inspection of the plant during operating conditions and appeared not to address management expectations relating to job site conditions during outages. The licensee acknowledged the inspectors comment.

Issue #2: The team noted that system engineer required walkdown inspections did not cover observations/examinations of structural items. Station procedure SI-SG.1010, "Systems Engineering Walkdown and Area Inspection Guidelines," was subsequently revised to include a walkdown inspection checklist devoted entirely to structural items.

Issue #3: The team identified one concern regarding quarterly trend analysis reports not being issued in a timely fashion. During this inspection it was determined that there had been some improvement in the timely issuance of these reports. Quality Assurance Department Procedure 16.02, "Trend Analysis," requires that trend analysis reports shall be prepared and issued within forty-five days after the end of each calendar quarter. The inspector noted that although there had been some improvement in the timely issuance of reports they were still not all being issued within the procedurally required forty-five days. The licensee acknowledged the inspector observation and is assessing appropriate corrective actions.

Issue #4: The team noted that in the screenhouse the salt service water pump rooms had piping and supports with excessive amounts of rust and that grounding straps on several salt service water pump motors were corroded. In response to these observations, rusted supports have been cleaned or replaced and coated with an epoxy coating to inhibit future rusting. Also, the corroded grounding straps have been repaired. In addition, the licensee plans to overhaul the traveling screen, install new chlorination and cathodic protection systems and new fiberglass floor gratings during this outage.

Issue #5: The team identified other deficiencies in the screenhouse dealing with a breached flood control flapper valve fire barrier, and broken springs in fire dampers located within ventilation ducting. The licensee actions to correct these deficiencies are described in detail in Licensee Event Report 90-019-00. Briefly, the corrective action consisted of refurbishing the flood control flapper valve to reestablish the fire barrier and replacing the broken damper springs with new springs. An increased damper surveillance frequency will be maintained until springs made of a material better suited to the harsh pumphouse environment become available.

Issue #6: The team identified an inadequate method of storage for large metal clad gaskets in the warehouse area. In response to this concern, the licensee constructed three racks for the storage of large flat materials. These racks are being utilized and have eliminated the problem. The overall appearance of the warehouse was observed to be very good. Materials staged for specific work activities were clearly identified, segregated, and neatly stored.

Issue #7: The team determined that no job specific training program existed for maintenance planners. Also, the first line maintenance supervisory personnel training program was only in the developmental stage. In addition, it was noted several first line supervisors had no specific procedural training prior to assuming their positions. Currently, training for planners is under development with planner training scheduled to begin in September. Maintenance supervisor training requirements have been included in the maintenance section manual. These training requirements mandate receipt of both procedure training and supervisor orientation training prior to personnel assuming supervisory positions.

Issue #8: The team expressed concern with the high turnover rate experienced among the first line maintenance supervisory positions. During this inspection it was determined that since the MTI there has been very little turnover among first line supervisors. The most significant change has been in the electrical maintenance division managers position. Also, two supervisor positions vacant during the MTI have been filled.

Issue #9: As a result of a September 2-3, 1990 loss of feedwater control event, the licensee established a Multi-Disciplinary Analysis Team (MDAT). The results of the MDAT review were a number of recommendations. To assure incorporation of all MDAT recommendations into the maintenance program the Operations Review Committee (ORC) specifically tracked their disposition. The team felt the ORC involvement was a positive initiative. During this inspection the results of the ORC actions were reviewed. The ORC is tracking a total of 48 specific items resulting from the MDAT. Of these 48 items, 30 have been closed, 16 have been evaluated by the ORC, and two remain open.

Issue #10: The team determined during the review of MDAT findings that current controls make the completion of minor repairs a very time consuming activity. This was not an MDAT observation. However, the licensee has taken steps to help improve this condition by adding the Work Request Tag (WRT) to procedure 1.5.3, Maintenance Request. One of the functions of the WRT is to permit with certain controls, the completion and documentation of minor work on non-safety components on the WRT itself. This process serves to expedite the completion of minor work items.

Issue #11: The licensee performed a detailed review of the work control process. This review of Organizational Analysis and Refinement (OAR) was evaluated during the MTI and found to contain many good recommendations, some of which had been implemented even though the report had not been issued. The team felt the completion of the OAR could have been more timely in order to expedite the implementation of the report recommendations. During this inspection it was determined the OAR had been issued and that each recommendation not yet implemented was being tracked as an action item and a person responsible for its implementation had been identified.

4.1.4 MTI Followup Conclusion

The inspector concluded that the licensee has been providing timely resolution to MTI issues. A plant material condition inspection program has been implemented. However, this program does not appear to address outage conditions. System engineer walkdowns now include observation of structural items. Improvement has been noted in the timely issuance of QA Quarterly trend Analysis Reports. Further improvement here is still warranted. Improvements have been made in the screenhouse. Further improvements here and in the auxiliary bay area should be initiated through the new material condition inspection program. Improvement was noted in the warehouse. A planner training program is under development and maintenance supervisor training requirements have been specified. First line maintenance supervisor turnover has been reduced. The disposition of MDAT recommendations is being tracked by the ORC. A method for expediting minor maintenance has been established and OAR recommendations are being factored into the maintenance process.

4.2 Inadvertent ESF Actuations During Surveillance Testing

4.2.1 Reactor Building Isolation System "A" Train Actuation

On May 24, 1991 while backing out of a station battery surveillance procedure, an inadvertent actuation of "A" train of the reactor building isolation system (RBIS) was experienced. The RBIS caused automatic closure of the reactor building "A" train supply and exhaust ventilation dampers and the automatic start of the "A" train standby gas treatment system (SGTS) fan.

Licensee maintenance personnel were restoring the 125 Vdc system to normal alignment following the premature conclusion of a surveillance test when the actuation occurred. The surveillance, an eight hour discharge test of the "A" 125 Vdc battery, was aborted after approximately 30 minutes due to difficulties with the test equipment. The test electrical configuration cross connects the "A" 125 Vdc bus to the "B" 125 Vdc bus to facilitate isolation of the "A" 125 Vdc battery from the bus to conduct the discharge capacity. In order to properly restore normal 125 Vdc system alignment, the "A" battery should be restored and connected to the bus, then the "A" battery charger should be connected, and then the circuit breaker providing cross connect to the "B" 125 Vdc power supply and bus should be opened. However, while restoring the "A" 125 Vdc alignment following abort of the surveillance test, maintenance technicians improperly sequenced activities by connecting the "A" battery charger first, then opening the circuit breaker to the "B" 125 Vdc bus, and then lastly connecting the "A" 125 Vdc battery to the bus. The improper restoration sequencing caused erratic "A" battery charger operation. Specifically, 125 Vdc power was momentarily interrupted to distribution panel D4 which provides control and logic circuit power to the "A" train of RBIS thereby causing the actuation. The RBIS was reset and affected components were restored to normal configuration within six minutes. All subsystem components responded to the RBIS actuation as designed.

The root cause of partial RBIS actuation was personnel error in the restoration of the "A" 125 Vdc train to normal alignment. The surveillance procedure (8.9.8, "Station Battery Acceptance, Performance or Service Test," revision 23) developed sufficient instruction to properly complete restoration. However, the procedure is scheduled to be revised to include addition of caution statements regarding proper restoration sequence. Additionally, the involved individuals were counseled on the event and on the need to review and understand procedural direction.

4.2.2 Primary and Secondary Containment Isolation System Actuation

On June 4, 1991, an inadvertent actuation of the primary containment isolation system (PCIS) and the RBIS occurred during surveillance testing on a reactor protection system (RPS) motor-generator set electrical protection assembly (EPA). Specifically, the "B" RPS set EPA-3 was being functionally tested. Previous to initiation of the test, RPS channel "A" and PCIS channel "A" trip signal were present due to an unrelated maintenance activity on an RPS channel "A" low water level relay. The functional test of EPA-3 required that the EPA control switch be repositioned from the normal position to the test position. This test step caused the designed trip of EPA-3 and loss of 120 Vac power to panel C-511 bus "B". However, with the maintenance related RPS and PCIS channel "A" trip signal present, trip of the EPA-3 caused coincident channel "A" and channel "B" low water level trip signals to exist which resulted in the PCIS and RBIS actuations. The PCIS and RBIS responded to the actuation signal as designed. Both systems were reset within thirty minutes of the actuations.

The root cause of the PCIS and RBIS actuations was inadequate nuclear watch engineer review of plant conditions, specifically RPS status, prior to authorization to perform the EPA functional test procedure. At the time of the event, the reactor vessel was defueled with no core alterations in progress and the RPS, PCIS, and RBIS were not required to be operable.

4.2.3 Inadvertent Automatic Start of the "B" Emergency Diesel Generator

On June 7, 1991 during conduct of a diesel generator logic surveillance procedure, the "A" EDG inadvertently automatically started. The diesel started and attained rated speed as designed but did not provide power to its associated 4160 V bus (A-5) because the output breaker was deenergized and racked down as a test requirement.

The root cause of the inadvertent automatic start of the "A" EDG was personnel error during system restoration following surveillance test completion. Maintenance personnel relanded a lifted A-5 bus auxiliary relay (172-504X) lead before the startup transformer (152-504) or the unit auxiliary transformer (152-505) feeder breakers were racked in or closed. The logic configuration established caused the EDG emergency start relay to energize and the "A" EDG to automatically start. The diesel and normal electrical loads were isolated from the A-5 bus, therefore start of the EDG was of no consequence to the onsite electrical distribution system.

The three inadvertent ESF actuations documented above were separate and unique events. Each event was appropriately reported to the NRC Operations Center in accordance with 10 CFR 50.72 requirements. Although the root cause of each event w s attributable to personnel error, the events in the aggregate were not indicative of surveillance raining program weaknesses. These events were of minimal personnel or equipment safety impact. Throughout the conduct of the current outage, noteworthy licensee interdepartmental communications, coordination, and supervisory controls have managed critical path schedules and plant conditions such that inadvertent ESF actuations and personnel error related e nts have been effectively minimized. The inspector had no concerns regarding these events.

5.0 EMERGENCY PREPAREDNESS (40500)

5.1 Followup of Previously Identified Items

(Closed) Unresolved Item 50-293/91-02-01. Evaluation of licensee actions to resolve the delineation of communications responsibilities with offsite emergency organizations. During the January 11, 1991, smoldering turbine building roof event a thirteen minute delay was experienced between the time the nuclear watch engineer directed offsite firefighting assistance be requested to assess roof condition and the time the request was actually accomplished. Additionally, previous communications coordination difficulties had been experienced during response to an onsite medical emergency as well as during a medical emergency drill.

In response to these events the licensee revised existing station procedure 5.5.1, "General Fire Procedure," (current revision 13) to ensure a senior reactor operator (SRO) in the control room initiates notification to the Town of Plymouth Fire Department after the station fire brigade is dispatched to a fire. Additionally, new procedure, 5.5.3 "Medical Emergency Response Procedure," revision 0, was issued which directed that an SRO in the control room initiate notifications for offsite medical assistance if needed during medical emergencies.

Inspector review of the procedures, determined necessary responsibility and instruction have been established to provide assurance that timely notification to offsite emergency support organizations will be accomplished. This item is closed.

6.0 **SECURITY** (71707)

Selected aspects of plant physical security were reviewed during regular and backshift hours to verify controls were in accordance with the security plan and implementing procedures. This review included security measures; vital and protected area barrier integrity, maintenance of isolation zones, and implementation of access control including access authorization and badge issue, searches of personnel, packages and vehicles, and escorting of visitors. No discrepancies were noted.

7.0 **SAFETY ASSESSMENT AND QUALITY VERIFICATION** (90712)

7.1 **Licensee Event Report (LER) Review**

7.1.1 **LER 91-05**

LER 91-05, "Loss of AC Power to "B" Trains of Safety Systems Due to Diesel Generator "B" Voltage Regulator Failure During Surveillance," describes the March 25-26, 1991 loss of the 4160V A-6 bus event. The event is documented in NRC Inspection Report 50-293/91-04. The LER accurately recounted the event, cause contributions, and corrective actions. The inspector had no questions regarding this LER.

7.1.2 **LER 91-06**

LER 91-06, "HPCI and RCIC Systems Became Inoperable Due to Tripped Inverters," describes the March 26, 1991 HPCI and RCIC inoperability for a period of approximately nine minutes following a trip of the associated 125 Vdc inverters. The inverters tripped due to a voltage transient incurred during restart of the "B" recirculation pump. The event is documented in NRC Inspection Report 50-293/91-04. An Unresolved Item (50-293/91-04-01) was issued to review and assess the licensee evaluation and corrective actions to this event. The LER appropriately addressed the reporting criteria and included discussion of the delayed 10 CFR 50.72 required Emergency Notification System (ENS) report.

Currently, the licensee is reviewing a plant design change (PDC) to adjust the inverter trip setpoints such that voltage transients resultant from routine start of recirculation pumps or similar electrical loads will not effect inverter operation. The inspector will track and review PDC development and implementation via the open Unresolved Item. This LER is closed.

7.1.3 **LER 91-07**

LER 91-07, "Completion of a Shutdown Due to Drywell Floor Sump Leakage Rate and Subsequent Scram Signal While Shutdown," describes the April 29-30, 1991 plant shutdown due to failure of the "B" recirculation pump seal package. Additionally, the report documented an unanticipated reactor protection system actuation during reset of the scram discharge instrument volume high water level scram bypass switch. These events are documented in NRC Inspection Report 50-293/91-04. The LER effectively detailed initiation of plant shutdown, increased unidentified leakage rates, Notification of Unusual Event declaration and termination, anticipated actuations of engineered safety feature systems, and entry into cold shutdown. The inadvertent RPS actuation was the result of reactor operator error during reset of the scram discharge instrument volume (SDIV) high water level scram bypass switch. The operator failed to ensure the scram signal had cleared after the SDIV was drained prior to resetting the bypass switch. As corrective actions, the reactor scram procedure was revised to require senior reactor operator verification that the SDIV high water level indications have cleared before reset of the bypass switch to normal. Additionally, the SDIV level indication circuitry is being evaluated by the licensee to determine if potential human factors enhancements may exist. The inspector had no questions regarding this event. This LER is closed.

7.1.4 **LER 91-08**

LER 91-08, "Three Automatic Group I Isolations due to False High Reactor Water Level Signals While Shutdown," describes the primary containment isolation system (PCIS) automatic actuations during the April 30, 1991 reactor depressurization following reactor shutdown. The three automatic Group I PCIS actuations occurred with reactor pressure at less than 100 psi and were the result of spurious reactor vessel water level instrumentation transients. The PCIS actuations are documented in NRC Inspection Report 50-293/91-07. An Unresolved Item (50-293/91-07-01) was issued to review and assess the licensee evaluation and determinations regarding the isolations. The LER effectively developed plant conditions and responses to the isolations. However, the LER was required to be submitted before the licensee investigation of the events was complete. Therefore, the LER lacks root cause and corrective action determinations and will require a supplemental update report submittal. Inspector review of this issue and licensee implementation of corrective action will be tracked via the existing unresolved item. The inspector had no questions with this report. This LER is closed.

7.1.5 **LER 91-09**

LER 91-09, "Fire Barrier Found Breached in Intake Structure," describes the May 18, 1991 licensee observation that the east wall fire barrier of the "B" train salt service water pump room located within the intake structure had been breached. A four inch room drain check or flapper valve on the east wall had been opened to allow an air hose to be passed through the penetration without appropriate compensatory measures being taken. Licensee investigation concluded the uncompensated condition existed for approximately one day. Upon discovery, a continuous firewatch was posted at the wall location. Additionally, similar fire barrier walls and penetrations in the intake structure were identified and labeled to preclude recurrence of a similar event. Licensee review of plant design bases documents and drawings concluded the drain check or flapper valves serve no apparent safety function. As such, the licensee took action to remove the valves and grout the openings thereby ensuring the integrity of the involved fire barriers.

7.1.6 **LER 91-10**

LER 91-10, "Inadvertent Secondary Containment System Isolation While Backing Out of Surveillance Testing Due to Personnel Error," describes the May 27, 1991 inadvertent actuation of reactor building isolation system train "A" while restoring normal 125 Vdc system configuration following an aborted surveillance test. The event is documented in Section 4.2.1 of this report. The LER appropriately addressed the reporting criteria. The inspector had no concern regarding this report. This LER is closed.

7.1.7 **LER 91-11**

LER 91-11, "Inadvertent Primary Containment and Secondary Containment Isolation Signal During Surveillance Testing While Shutdown for Refueling," describes the June 4, 1991 inadvertent actuation of the PCIS and RBIS while performing a functional test of a RPS motor generator set electrical protection assembly. The event is documented in section 4.2.2 of this report. The LER appropriately addressed the reporting criteria. The inspector had no concern regarding this report. This LER is closed.

7.1.8 **LER 91-12**

LER 91-12, "Automatic Start of Diesel Generator During Surveillance Testing due to Non-Licensed Personnel Error," describes the June 7, 1991 inadvertent automatic start of the "A" EDG during logic testing. The event is documented in section 4.2.3 of this report. The LER appropriately addressed the reporting criteria. The inspector had no concern regarding this report. This LER is closed.

8.0 ENGINEERING AND TECHNICAL SUPPORT (71707, 71710)

8.1 Followup of Previously Identified Items

(Closed) Unresolved Item No. 87-53-02.1. This item pertained to operator failure to adequately investigate alarms indicating a grid under-voltage condition and loss of RPS/PCIS analog trip systems cabinets during the November 12, 1987 loss of offsite power event.

In response to this event, the licensee implemented requalification training Module No. 88-0-RQ-08.01-07, Loss of Offsite Power Events of March 31, 1987 and November 12, 1987, dated April 28, 1988 to instruct operators on the potential consequences of failure to investigate annunciators. Additionally, the licensee included a loss of offsite power scenario in the simulator portions of the Licensee Operator/STA Requalification Training Program No. 88-0-RQ-08-01-08. The inspector determined the licensee actions were appropriate to effectively addressed this issue. This item is closed.

(Closed) Unresolved Items No. 87-53-01.3. This item pertained to the lack of procedural guidance for administratively staffing the Technical Support Center (TSC) in situations where Emergency Plan Activation is not appropriate.

In response to this issue, the licensee developed a Nuclear Organization Procedure (NOP)-88A2, "Non-Emergency Notification of BECo/PNPS Management", dated of June 19, 1989, which describes the mechanism by which appropriate management personnel will be informed and mobilized to respond to events which do not warrant activation of Emergency Response Organization. This item is closed.

(Closed) Unresolved Item No. 87-22-01. This item pertained to the licensee commitment to develop a document that clearly described the detailed bases for Appendix R compliance. The licensee completed report No. 89XM-1-ER-0, "Updated Fire Hazards Analysis," dated July 12, 1990, which documented the PNPS Fire Protection Program. Included in the document was Calculation No. PS-32, Revision 3, "Appendix R Safe Shutdown Analysis" dated October 1, 1990. This documented PNPS compliance with Appendix R requirements.

The report was a consolidation of information and analyses that have been prepared in response to Appendix A to the NRC Branch Technical Position 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976" and Appendix R to 10 CFR 50 as it related to fire protection. The report also referenced the Shutdown Analysis that was prepared to demonstrate the safe shutdown capability of PNPS in accordance with Appendix R requirements. Inspector review of the fire hazards analysis determined the report presented the information in a clear, traceable manner, with support documentation provided to ensure compliance with appropriate fire protection requirements. This item is closed.

(Update) UNR 50-293/91-07-01. The engineering department reviewed the mechanism by which reactor water level indication spiked during the plant cooldown on April 30, 1991, resulting in three automatic primary containment isolation control (PCIS) system Group 1 isolations. The mechanism appears to be that during reactor depressurization, flow patterns within the head equalizer line become disrupted due to steam formation and prevent open hydraulic communication between the reference leg and the reactor vessel. The head equalizer line runs between the reactor nozzle and the reference level condensing chambers. This disruption decreases pressure in the condensing pots with respect to the reactor causing a false high level spike. Discussion with vendor personnel and other licensees with similar instrument line designs indicated the one inch diameter equalizing line may be marginal in sustaining a single, stable flow pattern during reactor pressure changes. As corrective actions, the licensee has replaced the reactor water level equalizing lines between the reactor vessel nozzle and condensing chambers (train "A" and "B") with two inch piping. Post work testing to verify the adequacy of the licensee corrective action will be further evaluated by the inspector. This item remains open.

8.2 Salt Service Water Inspection and Piping Replacement

During ultrasonic thickness (UT) measurements of the Salt Service Water (SSW) system, a through wall pinhole leak developed on spool piece JF-29-15-5 (suction to SSW pump P-208D). The SSW system is the ultimate heat sink. This safety system is comprised of rubber lined carbon-steel piping which supplies sea water to the reactor building closed cooling water (RBCCW) and turbine building closed cooling water (TBCCW) heat exchangers during normal and emergency conditions. To confirm system operability, the licensee performed an evaluation utilizing the methodology provided in NRC generic letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping" as a technical basis for establishing the structural integrity of the piping. Although an ASME code repair was accomplished (i.e. spool piece replacement), the licensee concluded this condition was the result of localized pitting and did not impact piping structural integrity. The inspector concluded that this operability determination was appropriate, however due to the safety significance of the SSW system the inspector performed a detailed review of the SSW inservice inspection efforts and pipe replacement program.

The controlling document for the inspection plan was NED 91-145, "Salt Service Water Piping Routine Inspection," Revision 1. This plan was developed per the requirements of NRC generic letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The inspection program consisted of visual surveys of piping exterior surfaces, UT measurements, and visual examination of the rubber lining. Licensee sensitivity to the SSW system was evidenced by the performance of UT testing, which is beyond ASME section XI requirements for code class 3 piping. The inspector reviewed all UT results from the current inspection and observed the UT measurements of spool piece JF-29-4-5 per MR 19100465. Nonconformance reports (NCRs) were initiated for UT measurements less than minimum wall thickness and the effected piping was subsequently replaced. Additionally, during the visual inspection of spool piece JF-29-16-9, the licensee observed that the rubber lining was peeling off (delaminating) from a section of the pipe. Rubber lining delamination has the potential to result in either RBCCW heat exchanger fouling or accelerated corrosion of the exposed carbon steel piping. The licensee responsively initiated augmented inspection efforts and found no other areas of gross delamination. Inspector review of remote video inspection tapes of inaccessible portions of buried "A" loop SSW piping also revealed no areas of gross delamination. The inspector did note however the near total absence of marine growth, indicative of an extremely effective biofouling control program. The inspector concluded the licensee's inspection program is properly performing its function by identifying and correcting deficiencies before they degrade the capabilities of the SSW system.

In addition to an effective inspection program, the licensee is in the process of implementing a SSW piping replacement program. During the current refueling outage, 14 spool pieces were replaced with new carbon steel piping. However, four spool pieces were replaced without the rubber liner due to physical constraints during installation. The general corrosion rate of the unlined carbon steel pipe in a sea water environment will be accelerated. Based upon review of carbon steel corrosion rates in salt water and baseline UT measurements, the inspector concluded adequate corrosion margin exists prior to the pipe replacement during the 1992 mid-cycle outage. The replacement of the remaining buried and non-buried SSW piping is currently scheduled for completion during the 1993 refueling outage.

The inspector found the licensee SSW inspection program to be thorough, well detailed, and effectively implemented. Nuclear Engineering Department disposition of NCRs generated during the SSW piping inspection was comprehensive and well supported. Additionally, replacement of the 14 spool pieces was both aggressive and responsive to the concerns identified by the inspection program.

9.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (30703)

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 inspection period, the resident inspector staff conducted an exit meeting with licensee management summarizing inspection activity and findings for this report period. No proprietary information was identified as being included in the report.

9.2 Management Meetings and Other NRC Activities

On May 28, 1991 Representative Peter Kostmayer, Chairman of the House subcommittee on Energy and the Environment, and subcommittee staff members; Federal Emergency Management Agency, Region I Administrator and members of his staff; and Mr. William Kane, NRC Region I Deputy Regional Administrator and members of his staff, visited the site.

On June 3-7, a NRC Region I engineering specialist conducted an inspection of the licensee engineering and technical support programs. Inspection results will be documented in Inspection Report 50-293/91-16.

On June 3-7, a NRC Region I engineering specialist conducted an inspection of the licensee inservice inspection (ISI) activities during the refueling outage. Inspection results will be documented in Inspection Report 50-293/91-14.

On June 12, a public meeting was convened to receive comments on the draft report of the Pilgrim Offsite Emergency Preparedness Task Force.

On June 17-21, a NRC Region I health physics specialist conducted an inspection of the licensee contamination and internal exposure control programs. Inspection results will be documented in Inspection Report 50-293/91-13.

On June 25, NRC Management Meeting Number M-91-71 was convened at the Chiltonville Training Center to discuss licensee self-assessment initiatives. The licensee distributed the prepared overhead displays which are included as Attachment 1 to this report.

Boston Edison
Pilgrim Nuclear Power Station

Self Assessment

Introduction

G.W. Davis
Senior Vice President-Nuclear
June 25, 1991

Boston Edison
Pilgrim Nuclear Power Station

Self Assessment

Introduction

R.A. Anderson
Vice President
Nuclear Operations and
Station Director
June 25, 1991

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Self Assessment

Operations

L.J. Olivier
Deputy Plant Manager
June 25, 1991

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Operations Improvement Initiatives

- Computer generated tagout system improves efficiency, standardizes tagouts
- Operational administrative requirements procedure established, conservative system component requirements in addition to technical specifications
- RFO #8 operations and special considerations procedure implemented
- Plant tour procedure upgrade
- Operational procedure upgrade commitment completed
- Procedural adherence as a way of life
- Emergency Operating Procedures (EOP's) upgraded
- CCTV and robotics for monitoring equipment in high radiation areas
- LCO tracking program initiated for trending

Plant Performance Better Than Ever

- No scrams due to operator error
- 217 days continuous run (9/90 - 4/91)
- No automatic shutdowns
- Acknowledged exceptional operator performance

OPS Staff Highly Qualified

- All positions filled
- 3 SRO's per shift
- Shift Control Room Engineer program implemented
- Current staffing - 23 SRO's, 20 RO's, 27 Non-Licensed
- Present license class: 3 SRO, 6 RO candidates
- Next license class begins this fall
- Fifth straight license class with 100% pass rate

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Off Normal Events Demonstrate Superior Operator Performance

- Feedwater regulating valve malfunction - (9/90)
- Turbine building roof fire - (1/91)
- Loss of Bus A-6 - (3/91)
- B Recirculation Pump seal degradation - (4/91)

Operations Management Actively Involved In Day-To-Day Activities

- Direct SRO supervision for major evolutions
- Pre-Evolution briefings for major activities
- Specific SRO involvement in RFO #8 planning
- SRO review of system safety while in RFO #8

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We Are Determined To Be The Best

- Conservative plant operation
- Strong safety culture
- Commitment to excellence

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Pilgrim Nuclear Power Station

Self Assessment

Plant Maintenance

E.S. Kraft, Jr.
Plant Department Manager
June 25, 1991

Improvements On All Fronts In Maintenance

- Quality of workmanship
- Work control process
- Preventive maintenance

Improvements In Quality Of Workmanship Yield Results

- Workforce structured in teams
- Maintenance quality improvement program initiated
- Rework identification/reduction program focuses on root cause
- Balance of plant quality program is upgraded
- Quality of workmanship training developed
- Supervisor training under development

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Work Control Process Continues To Improve

- Organizational analysis, multidisciplined task force review work control
- Revised maintenance request procedure focuses on improved material condition
- Maintenance planning is integrated into MR process
- Improved adherence to procedures

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Preventive Maintenance Improvements Underway

- Predictive maintenance upgrades in thermography, oil analysis, vibration analysis complete

- HPCI and RCIC PMS underway, based on system improvement program

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Maintenance Staff Improvements

- Staffing near authorized complement
- First line supervisor staffing increased
- INPO peer evaluators add to experience
- Routine maintenance performed by BECo personnel - no contractors

Plant Material Condition Good, Improving

- Open power block MRS low
- Significant upgrades during RFO #8
  - Screenhouse repairs implemented
  - SSW piping replacement in progress
  - Extraction steam, moisture separator piping replaced
  - RWCU piping IGSCC susceptible material being replaced
- Annual NEIL and ANI inspections rate Pilgrim material condition higher than BWR average

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Pilgrim Nuclear Power Station

Self Assessment

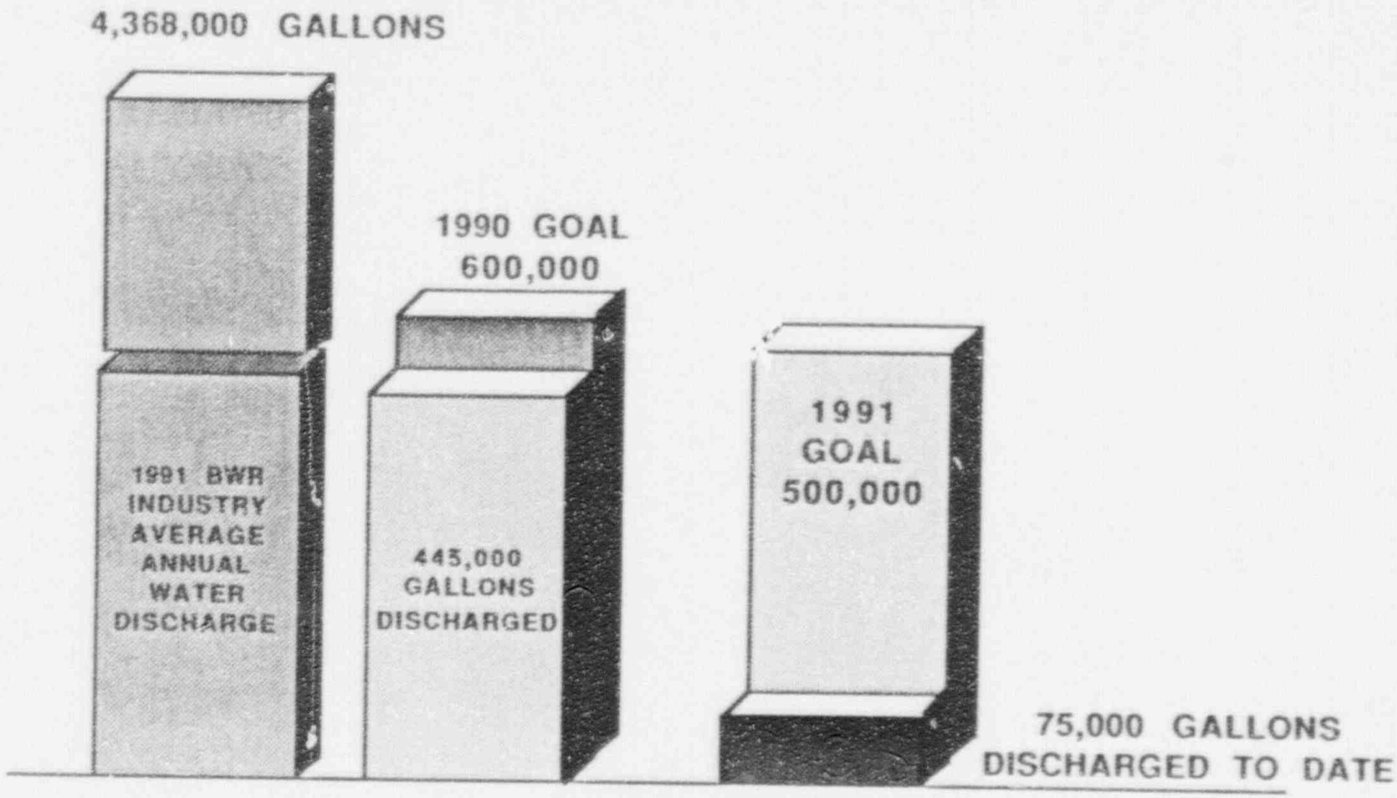
Radiological Controls

L.L. Schmeling
Radiological Controls
and Chemical Processes
Department Manager
June 25, 1991

Focus On Continuing Radiological Improvements

- Formation of radiological controls and chemical processes department combines radiological operations under one manager
- 1990 overboard liquid discharges only 10% of BWR average
- On-site hazardous materials and radioactive waste sharply reduced
- Programmatic improvements for postings, waste and chemical control and dose limit upgrade contribute to success
- Improvements help minimize exposure
- Increased supervision, management control of radiological operations

Processed Water Discharge



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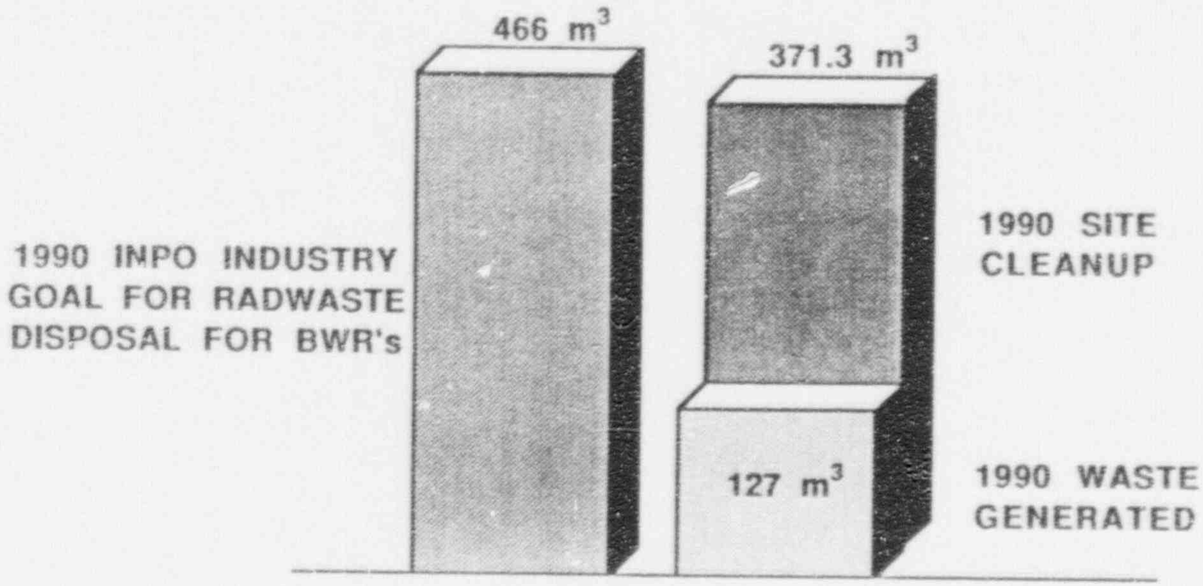
1990 Radwaste Site Cleanup



Specific Actions Taken To Improve Past Issues

- High radiation area door control improved
- Discharge calculation accuracy assured through proceduralized review
- QA surveillance of radwaste operations improve waste handling

Radiological Issues Resolved Swiftly, Positively

- Improved procedures, controls and supervision upgrade TCF operation

- Human factor improvements in postings and heightened awareness by plant personnel

- Vigilent management oversight minimizes radiological issues

High Quality Staff Makes A First Class Operation

- Key positions filled with experienced, well-qualified people with trained back-ups
- HP staff turnover stabilized
- Formalized experience and training standards for in-house and contractor HP technicians cited as excellent initiative
- INPO evaluations of HP, chemistry training notes program strengths

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Pilgrim Nuclear Power Station

Self Assessment

Security

D.J. Long
Plant Support Department
Manager
June 25, 1991

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Staffing Contributes To Program Success

- Highly qualified management
- Fully staffed
- Continuous oversight of guard force
- Overtime controls effective
- Limited use of Consultant Personnel

Quality Procedures Are A Program Strength

- Revised security plan
- No violations
- Security awareness by plant staff
- Training emphasizes adherence

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Security Plays An Active Role In Plant Maintenance

- Security represented on planning groups
- Review of plant design changes
- Temporary plant assignments
- Priority of security maintenance requests

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Significant Accomplishments Made This Period

- Annual LLEA response conference expanded
- Membership in the Southeast Mass. Detectives' Association
- Additional security awareness training
- Tactical response & firearms training
- Training to identify vital area boundaries

Future Enhancements Focus On Excellence

- Completion of the new security building
- Renovation of the lower main access point building
- New security diesel
- CCTV video capture system
- E-field upgrade
- Site lighting improvement

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Pilgrim's Security Program Strong, Improving

- Aggressive challenge to ensure system works
- Strong program
- Experienced/qualified personnel
- Modern technology
- Commitment to continued improvement

Boston Edison

Pilgrim Nuclear Power Station

Self Assessment

Engineering and Technical Support

R.V. Fairbank
Manager - Nuclear Engineering
June 25, 1991

Engineering And Technical Support Continues To Be An Organizational Strength

- Recognized strengths have been maintained
- Performance improves by implementing initiatives
- Achievements contribute to organization success

Recognized Strength Maintained

- Resources managed to meet long-term objectives
- Processes deliver results
- Continue to set and meet rising standards
- Engineering and Technical Staff is our greatest strength

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Engineering And Technical Staff Is Our Greatest Strength

- Staff technically self sufficient
- Low turnover
- Engineers attend SRO certification classes
- Technical seminars provide technology updates

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Performance Improves By Implementing Initiatives

- Increased day-to-day plant support
- Process changes increase effectiveness and efficiency
- Outage preparedness and support better than ever
- New engineering and analysis applied to plant support

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Achievements Contribute To Organization Success

- Improve plant availability
- Improve safety system availability
- Reduce personnel radiation exposure
- Improve industrial safety
- Improve support to operations
- Improve support to maintenance

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Pilgrim Nuclear Power Station

Self Assessment

Emergency Preparedness

R.A. Varley
Department Manager
Emergency Preparedness
June 25, 1991

Drills/Exercises Challenge The Organization

- Simulator usage began April 1991, will continue
- Drill scenarios developed on simulator
- Quarterly drills challenge emergency organization
- Personnel participation rotated to involve many

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Quality A Major Emphasis In EP

- Quality maintained high through:
  - Surveillance of drills/exercises
  - Prompt resolution of issues (1/11/91 UE notification)
  - Procedural improvements, controls
- Self assessment gives us frequent feedback:
  - Staff involvement at E.P. seminars
  - Participation in other utility exercises
  - In-house assessments
  - QA audits

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Staff And Facilities Fully Prepared For Emergencies

- EP department staff stable
- Emergency organization staff is at least 3 deep, most positions 4 deep
- Response staff trained, performance evaluated
- Qualifications and EP assignments tracked

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PNPS Offsite Program Progress

- Continued manpower funding, equipment support offsite program
- FEMA's recent finding of offsite program adequacy positive indicator
- NRC/FEMA task force report (NUREG 1438) - June 6, 1991
- MCDA action on other open items of concern accelerated

Boston Edison
Pilgrim Nuclear Power Station

Self Assessment

Safety Assessment And Quality Verification

R.A. Anderson
Vice President
Nuclear Operations and
Station Director
June 25, 1991

Response To NRC Observations

- Increased ORC involvement in site activities
- Computerized tagging system implemented
- Logging of lifted leads and jumpers improved
- Procedures upgrade program showing results

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Operating Management Pro-active In Assurance Of Safety

- Focus on redundant safety systems during RFO #8
- Management tours and inspections help assure quality, safety
- EPIC computer enhances operations, aids in root cause assessment
- New EOP's developed and implemented
- Nuclear Managers Committee oversees organizational operations
- New contractor examination and training standards

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Management Assures Thorough Corrective Actions

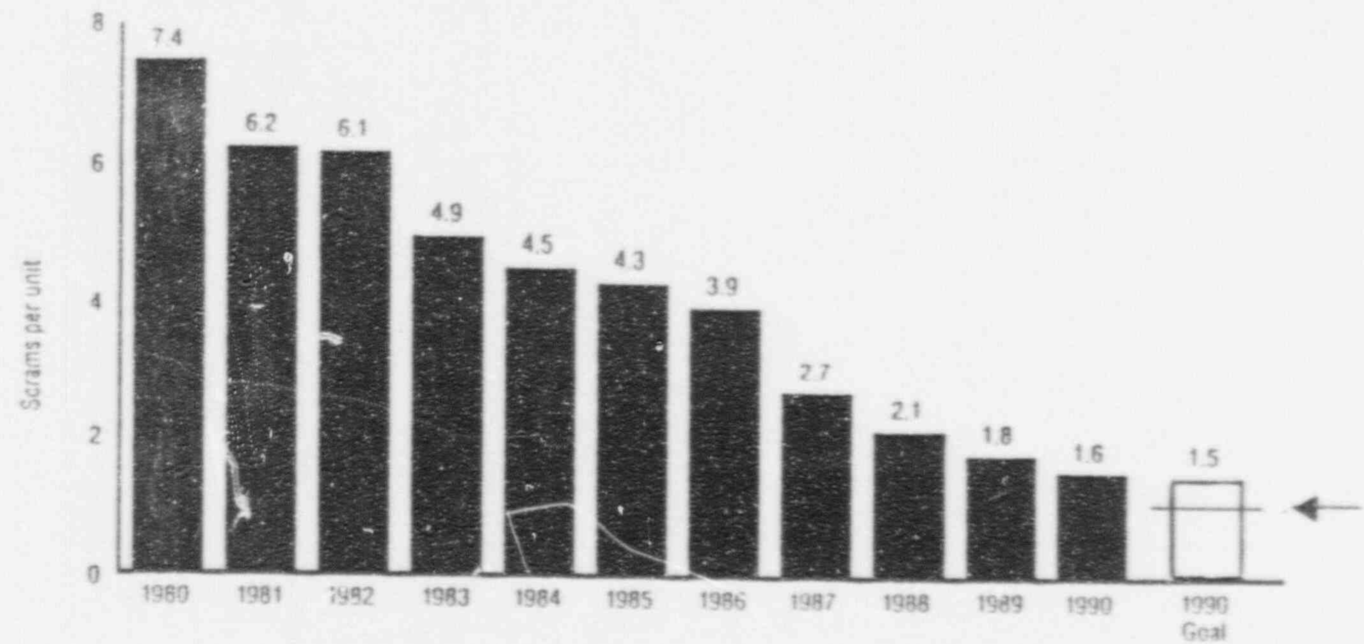
- Problem assessment committee analyzes corrective actions daily
- MDAT analyzes complex events
- Implemented HPCI/RCIC integrated improvement plan
- Upgraded trash compaction facility
- Design upgrade of valve 1001-50

BOSTON EDISON

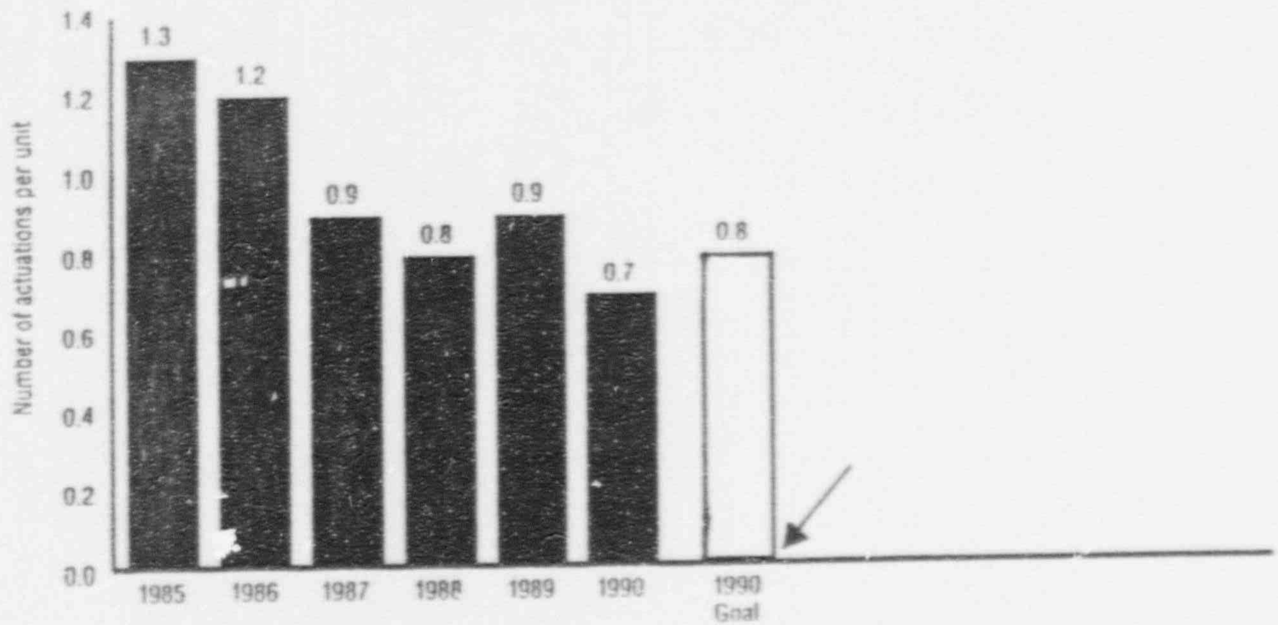
QA/QC Independent, Pro-Active And Respected

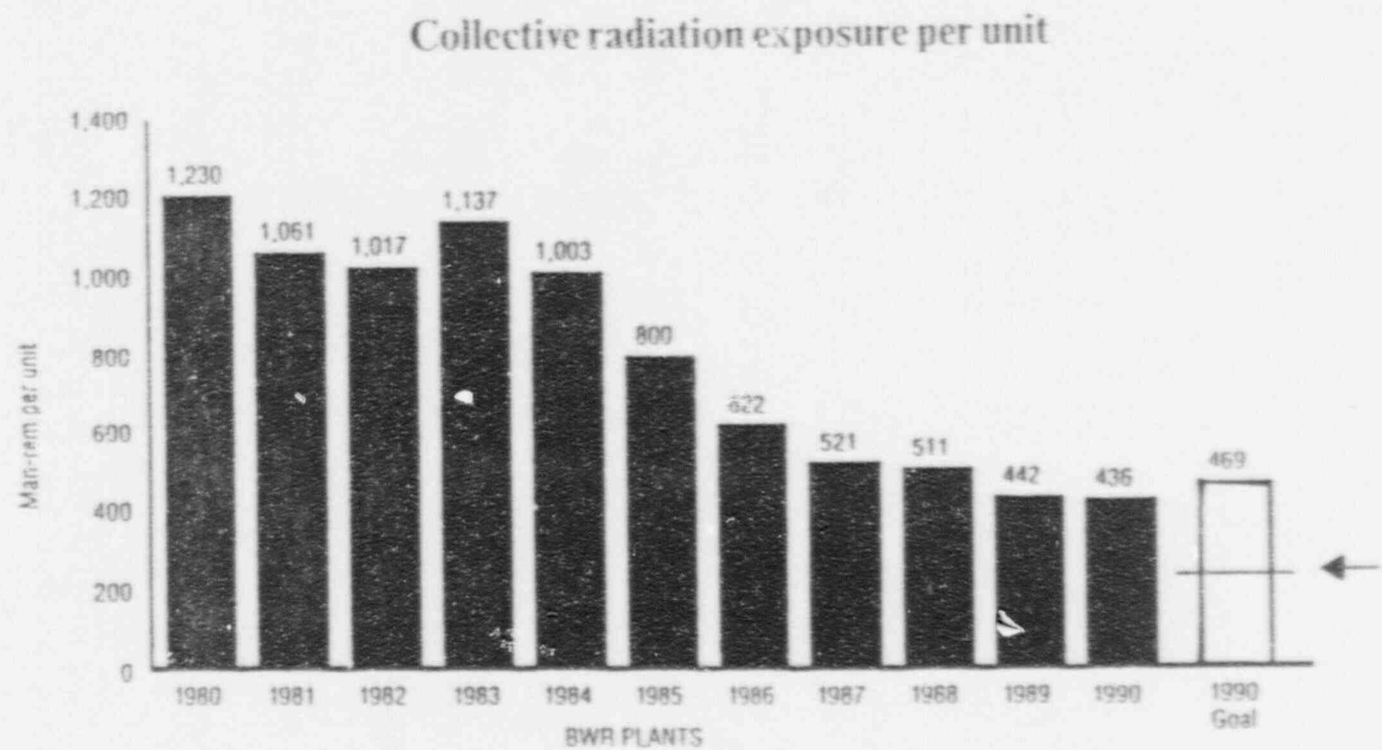
- Independent in-depth audits and surveillances
- QA manager on OPS Committee of NMC
- Operating managers request independent audits

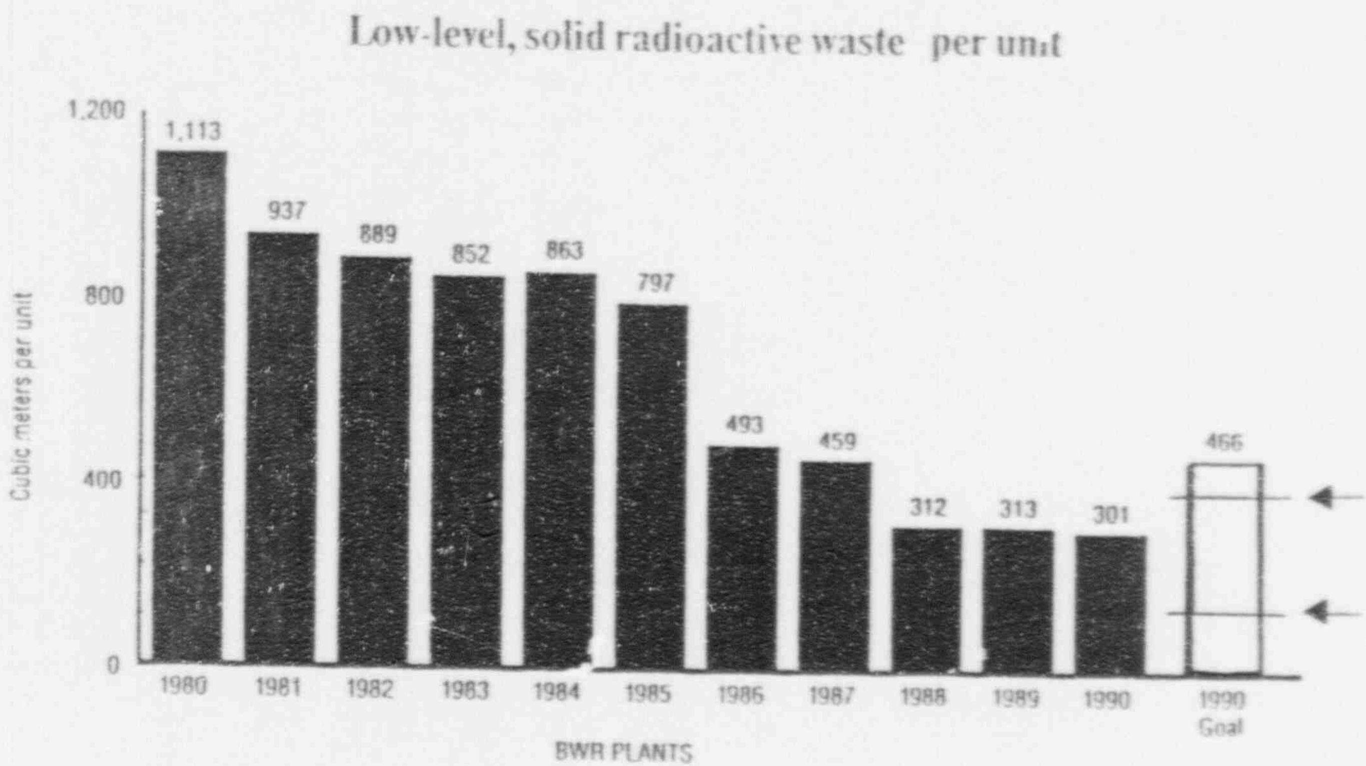
Unplanned automatic scrams



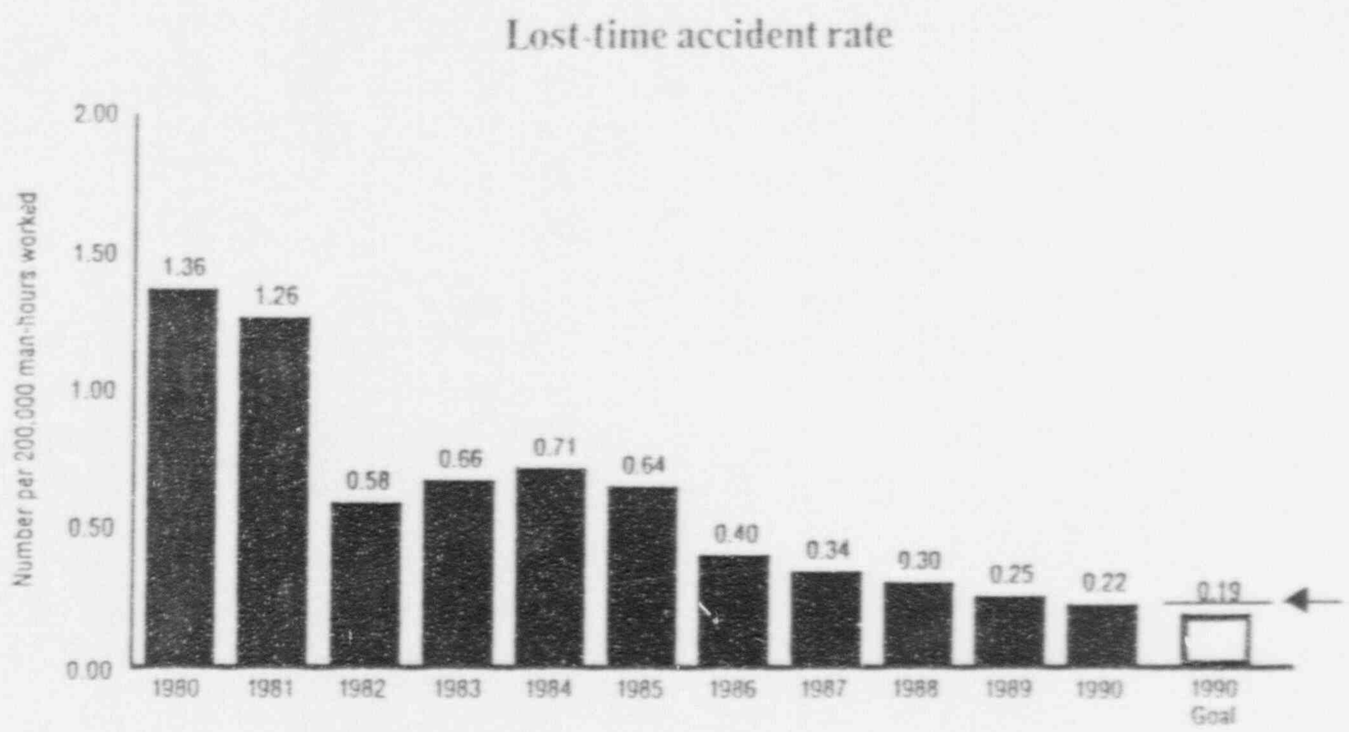
Unplanned safety system actuations



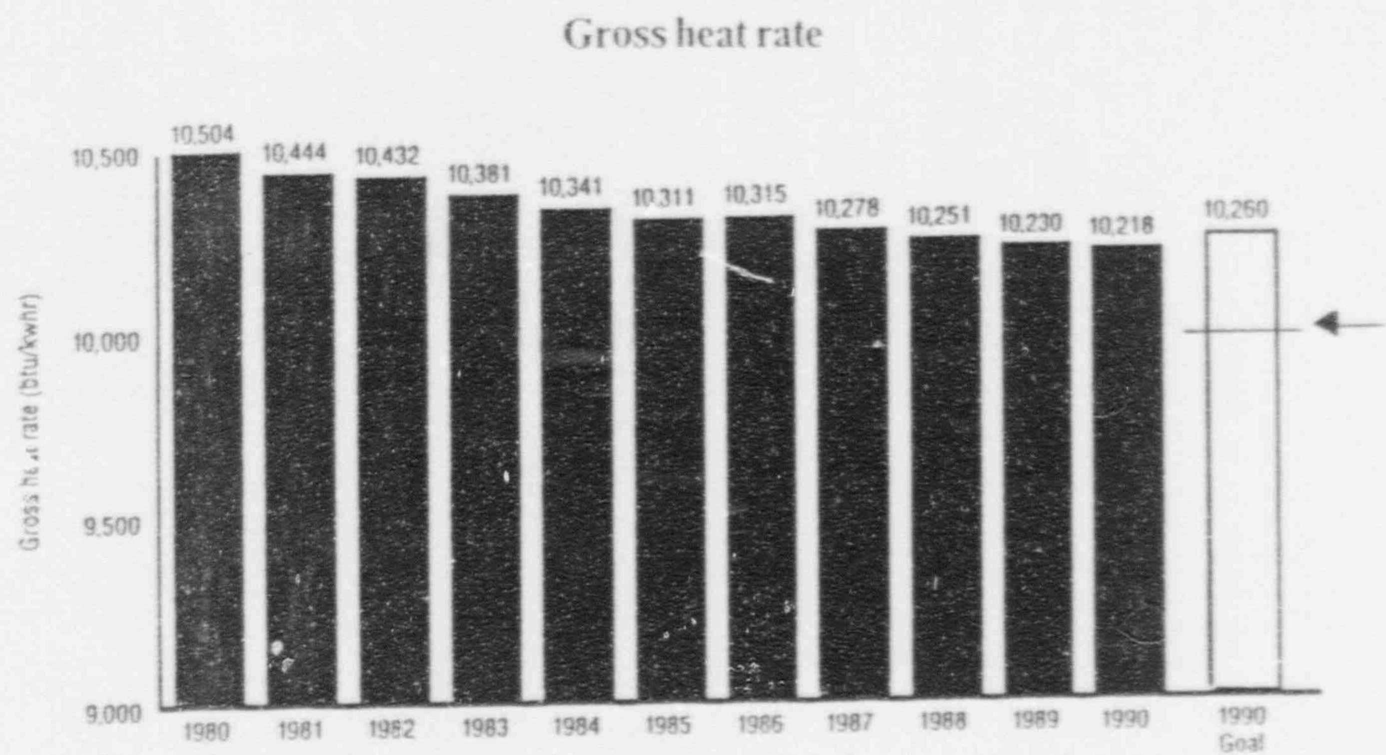

Collective radiation exposure per unit
1,400
1,200
1,000
800
600
400
200
0
Man-rem per unit
1,230
1,061
1,017
1,137
1,003
800
622
521
511
442
436
469
1980
1981
1982
1983
1984
1985
1986
1987
1988
1989
1990
1990 Goal
BWR PLANTS



BOSTON EDISON

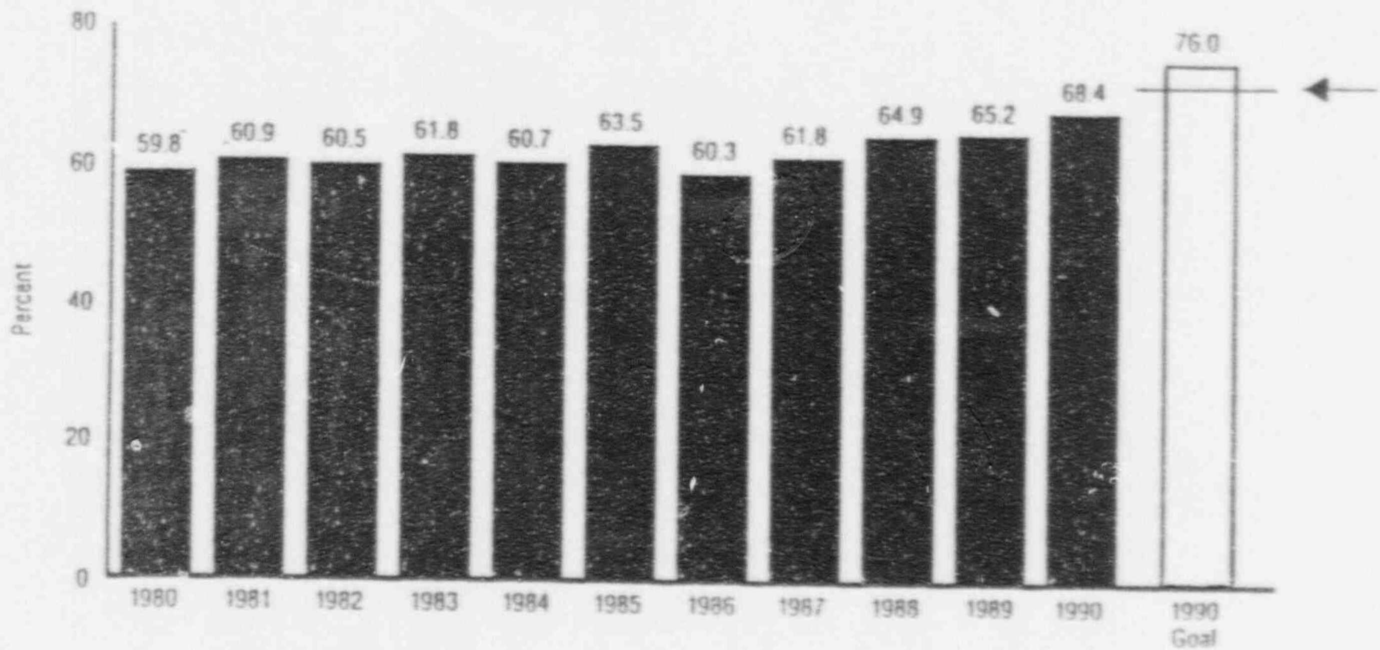


BOSTON EDISON



Gross heat rate
Gross heat rate (btu/kwhr)
10,500
10,000
9,500
9,000
10,504
10,444
10,432
10,381
10,341
10,311
10,315
10,278
10,251
10,230
10,218
10,260
1980
1981
1982
1983
1984
1985
1986
1987
1988
1989
1990
1990 Goal

Equivalent availability factor



BOSTON EDISON