U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 94-06

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

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: February 22 - April 4, 1994

Inspectors: J. Macdonald, Senior Resident Inspector
D. Kern, Resident Inspector

Approved by: R. J. Conte 4/25/94
R. J. Conte, Chief
Reactor Projects Section 3A
Date

Scope: Resident Inspector safety inspections were in the areas of plant operations, maintenance and surveillance, engineering, plant support, and safety assessment and quality verification. Initiatives selected for inspection included the review of operational and design configuration controls, forced outage activities, NRC Generic Letter 89-10 motor operated valve program status, security event log entries, and Licensee Event Report submittals.

Inspections were performed on backshifts during February 22-24, and 28, March 2, 4, 8-10, 15, 21, 23, and 24, 1994. Additionally, a deep backshift inspection was conducted on March 19, 1994 (12:00 noon - 2:35 pm).

Findings: Performance during this five week period is summarized in the Executive Summary.

EXECUTIVE SUMMARY

Pilgrim Inspection Report 94-06

Plant Operations: Control room operators maintained good control over plant parameters throughout the February 22, 1994, reactor shutdown and subsequent forced outage to repair a failed main steam isolation valve (MSIV). Operations section management conducted an extensive drywell inspection that identified two mispositioned feedwater system drain line valves. A review team assembled in response to this observation, as well as two other similar occurrences, was a positive initiative. The team composition established independence from normal operations functions. Ultimately, the review concluded positive configuration controls were being implemented. The review team was an appropriate initiative in light of the potential significance of the maintenance of safety system configurations.

Maintenance and Surveillance: The MSIV issue team established a comprehensive troubleshooting plan that addressed all potential contributing factors to the valve failure. Repairs and component replacements and post maintenance testing were properly implemented. Possible common cause inspections of the remaining MSIVs were conducted without discrepancy. Good efforts were accomplished to replace valves and fittings that had been injected with temporary leak sealant compounds during the operational cycle.

Engineering: Engineering personnel continued to effectively address motor operated valve performance issues. Of particular note was the investigation of a torque switch roll pin failure that resulted in the identification of a potentially significant generic sub-component deficiency that had been experienced previously in the industry but had not been formally reported to the NRC in accordance with the requirements of 10 CFR PART 21 criteria. Additionally, positive initiatives completed during the forced outage improved high pressure coolant injection system direct current motor and limit switch performance.

Plant Support: Security section supervision responded promptly to the notification by an onsite manager who was in receipt of a packaged bottle of wine. The subsequent investigation was thorough. The event was properly entered into 10 CFR PART 73.71 security logs, and the involved security force members were retrained and counselled on package inspection requirements. Soil samples obtained from locations around the new support building were determined to be representative of the overall soil present in the general area. Gamma spectrometry analysis of the soil samples when counted to a lower limit of detection indicated the presence of naturally occurring radioactive materials only. Independent NRC reviews concluded the test methodologies were appropriate.

Safety Assessment and Quality Verification: Licensee Event Reports continue to be of high quality. Event descriptions and corrective actions were accurate and complete. Quality assurance department representation on the configuration control review team provided objective independence from normal operational line functions.

TABLE OF CONTENTS

DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

At the start of the report period Pilgrim Nuclear Power Station was operating at approximately 80% of rated power, with the 'D' main steam line isolated due to the previous failure of the inboard main steam isolation valve (MSIV) to satisfy fast closure time requirements during routine surveillance testing. On February 22, 1994, at 11:00 pm, a reactor shutdown was initiated to facilitate troubleshooting and repair of the inoperable MSIV. The four way main control valve on the control air manifold to the MSIV was replaced due to the buildup of aluminum oxide on an internal booster piston. The MSIV was subsequently tested satisfactorily and declared operable (section 3.1). On February 26, 1994, during restoration of the feedwater system in preparation for plant restart, operators observed that the stem of non-safety related feedwater block valve, MO-3479, had separated from the disk and the valve was failed in the closed position. Repairs to MO-3479 were implemented and other startup prerequisites were completed on March 2, 1994.

On March 3, 1994, at 9:52 am, a reactor startup was initiated. The turbine generator was synchronized to the offsite distribution system at 10:57 am the following day and on March 5, 1994, at approximately 11:00 am the reactor achieved 100% of rated power. On March 9, 1994, the motor operator for the high pressure coolant injection system (HPCI) injection valve, MO-2301-9, failed in the closed position during surveillance testing. The valve is normally open and is closed to maintain pressure isolation during stroke time testing of the normally closed downstream injection valve, MO-2301-8. The HPCI-9 valve was opened by hand, the motor actuator was removed, and the valve was clamped in the open position pending analysis of the failure mode (section 4.4). On March 11, 1994, reactor power was decreased to approximately 45% due to increasing augmented off gas system (AOG) back pressure during removal of the AOG pre-filters. The AOG system had been operating at higher back pressures since startup from the forced outage due to water and moisture collection within the system. Reactor power was further reduced to approximately 30% to facilitate maintenance and water removal from the AOG. On March 12, 1994, reactor power was decreased to approximately 14% and the turbine generator was secured to isolate a pin hole leak that was identified on a stator water cooling line instrument connection. The instrument connection was capped and reactor power ascension was initiated. The turbine generator was synchronized to the offsite distribution system at 10:55 am, on March 13, 1994, and the reactor achieved 100% of rated power at 10:10 pm that evening.

The reactor remained at 100% through the conclusion of the report period. The HPCI-9 valve remained clamped in the open position, with the motor operator removed, maintaining the HPCI system in an operable standby condition.

2.0 PLANT OPERATIONS (71707, 40500, 71714, 90701)

2.1 Plant Operations Review

The inspector observed the safe conduct of plant operations (during regular and backshift hours) in the following areas:

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

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

Plant housekeeping, including the control of flammable and other hazardous materials, was 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.

2.2 Configuration Control

During this inspection report period, the licensee identified three instances in which hand operated valves were in an off normal position without apparent proper supporting administrative configuration controls. Specifically; two feedwater system drain line isolation valves were found in the open position while the system was out of service to repair feedwater block valve MO-3479; series residual heat removal system valve 10-HO-501B was found in the open position and 10-HO-502B was found in the closed position, normal configuration would call for the opposite position for each valve; and circulating water system scavenging valves 27-HO-1,2 5,&6 were found to be in the open position following conclusion of a main condenser backwash. In each instance, the actual configuration did not effect associated system operability and basic intended system functions were maintained. Notwithstanding, operations section management initiated a multi-disciplinary review of configuration control processes to determine if the identified problems were symptomatic of programmatic or performance weaknesses. The review team consisted of personnel from operations support, compliance, technical programs, engineering, and quality assurance. The initial review concluded the feedwater drain line isolation valves were within the maintenance boundary established for the MO-3479 repair and would have been included in the final feedwater system line up verification following completion of the maintenance. Additionally, the RHR valve positions had been evaluated and documented on system line up records however, shift supervision did not initiate an administrative process to document the actual configuration. Finally, the scavenging valve mispositioning was the result of miscommunication between operations personnel during the condenser backwash.

Extended review team activity consisted of a complete audit of system line up records. Additionally, all accessible portions of six plant systems were physically walked down. Three additional configuration discrepancies were identified by the review team. Again, in each instance the actual as found configurations did not effect system operation or function. Specifically, a carbon dioxide fire protection system vendor fill valve was in the open position; RHR sensing line root valve, 10-HO-20B, was documented to be in the open position without appropriate administrative control; and control rod drive sample line isolation valve was found in the open position due to weak chemistry procedure controls.

The review team concluded the occurrences of improperly positioned valves were due to lack of supervisory oversight to ensure configuration control procedures are properly implemented. Continued barrier analysis concluded that there were no significant contributing factors to the events.

The inspector discussed each occurrence with operations section management, reviewed the final review team report, and independently reviewed potential system and plant impact and similarly concluded each event was of minimal significance. Nonetheless, the review team initiative was appropriate and necessary to ensure programmatic configuration control weaknesses did not exist. Additionally, the team composition ensured a level of process independence. The review was comprehensive and used accepted causal and barrier analysis methodologies. The inspector concluded that the review verified that overall system configuration control was being properly maintained.

2.3 Outage Operations

The inspector observed various portions of the preparation and conduct of the plant shutdown to troubleshoot and repair the inoperable main steam isolation valve. The forced outage schedule effectively integrated maintenance activities with requisite plant conditions. The initiation of drywell de-inerting was properly briefed and implemented consistent with Technical Specification requirements. Reactor shutdown and entry into shutdown cooling system operations were similarly well controlled. The inspector accompanied the chief operating engineer (COE) on a detailed drywell inspection during which the COE identified that the feedwater system drain line isolation valves discussed above were in the off normal open position without apparent appropriate administrative controls. Overall, drywell housekeeping was very good. Areas in the vicinity of the MSIV repairs were observed to be clean and restored to normal conditions. Reactor startup, following repairs to the MSIV and emergent repair of feedwater system block valve MO-3479, was also well controlled. The inspector had no questions regarding conduct of operations throughout the outage evolution.

3.0 MAINTENANCE AND SURVEILLANCE (61726, 62703, 71710, 90712)

3.1 Main Steam Isolation Valve Repair

On February 17, 1994, the 'D' main steam line inboard main steam isolation valve (MSIV), AO-203-1D, failed a quarterly full fast closure test. Subsequently, the 'D" main steam line outboard MSIV was controlled closed and the steam line was isolated in accordance with Technical Specification requirements (TS 3.7.A.2.b). The test failure was previously documented in NRC Inspection Report 50-293/94-02, Section 2.4.

On February 22-23, 1994, a reactor shutdown was conducted to troubleshoot and repair the valve. Prior to the shutdown, the MSIV had been stroke time tested on February 18, 19, and 21, 1994, with closure times of 4.02, 3.71, and 3.90 seconds respectively. Surveillance test acceptance criteria is for each MSIV to close within 3.0 to 5.0 seconds. After the drywell was de-inerted during the shutdown, the valve was inspected and exercised with no abnormal visual indications identified.

Following the shutdown, the pneumatic control manifold for the MSIV was disassembled and inspected. The manifold consists of three actuating solenoid valves, a four way main air control valve, a three way exercise control valve, a two way redundant exhaust valve, and associated pneumatic tubing. The solenoid valves were moving freely and determined to be functioning properly. However, a buildup of a moist substance, believed to be aluminum oxide, was identified on surface area of a booster piston in the four way main control valve. The four way valve is comprised of two double acting pistons that supply or vent air in response to MSIV control signals to open or close the valve. The suspect booster piston opens when pilot air (Nitrogen) is applied to the top piston, as occurs when the MSIV receives an open signal. The booster piston remains open as long as the MSIV open signal is present. Upon receipt of an MSIV fast closure signal, the pilot air is vented through the associated (#1 and #2) solenoid valves and the booster piston closes. As the booster piston closes, the bottom piston opens, providing supply air to the top of the MSIV cylinder and supply air to the top piston of the second booster piston, causing the second booster piston to open and creating a vent path for the air under the MSIV cylinder, and the MSIV rapidly closes. Licensee causal analysis concluded the aluminum oxide buildup was of sufficient magnitude to prevent the suspect booster piston from repositioning to the closed position upon receipt of an MSIV fast closure signal, as designed, which in turn prevented the remainder of the valve from actuating and closing the MSIV during the initial surveillance test. Further, the licensee concluded that system vibrations during subsequent attempts to close the valve were sufficient to free the booster piston which in turn allowed the MSIV to function properly during later closure strokes.

The four way main air control valve and the two way redundant exhaust valve were replaced on the effected MSIV. Additionally, the licensee inspected the four way main control valve booster pistons on six of the seven remaining MSIVs. The four way valve on the 'B' main steamline outboard MSIV was not inspected because it had been replaced in August 1993. The presence of minute amounts of aluminum oxide were wiped cleaned from two of the remaining MSIVs

(AO-203-1A and AO-203-2C). The origin of moisture to support formation of aluminum oxide is under review. The supply nitrogen is from liquid nitrogen expansion and should be of very high quality. Additionally, the licensee is developing a preventive maintenance schedule for inspection of the MSIV air control system at an increased frequency, consistent with the objectives of the reliability centered maintenance program.

The inspector independently reviewed the problem report (PR 94.9070), plant and vendor drawings, discussed the causal analysis with the issue team, and visually inspected the control air manifold during the drywell inspection. The inspector concluded licensee actions were appropriate and the causal analysis and corrective actions were thorough. The inspector had no further questions.

3.2 Outage Maintenance Summary

In addition to the repair of the MSIV, the licensee completed numerous other minor maintenance tasks that necessitate plant shutdown conditions to accomplish. Several valves and fittings that had been injected with temporary leak sealant materials during the operational cycle were replaced or repaired. Specifically, the 'B' outboard feedwater check valve body drain line isolation valve HO-200B that had been temporarily leak repaired on January 8, 1994, by leak sealant injection was replaced (NRC Inspection Report 50-293/93-23). Also, the blank flanges on the 'A' and 'B' train number two feedwater heaters that were temporarily leak repaired in November 1993, by clamp installation and leak sealant injection were replaced (NRC Inspection Report 50-293/93-20). Additionally, feedwater system drain valve HO-200B, that was identified to be leaking very slightly during the initial at pressure drywell entry, was replaced. The inspector reviewed portions of the associated work packages, disposition of post installation non-conformance reports, and observed portions of the repair activities. The inspector concluded the work was properly planned and effectively implemented.

4.0 ENGINEERING (37828, 71707, 92700, 92701)

4.1 DC Motor Replacement to Improve Motor Operated Valve Operability Margin

NRC Generic Letter 89-10 and its supplements requests licensees to verify the operability of safety related motor operated valves (MOVs) under design basis conditions. Calculations to support operability evaluations incorporate conservative margins to allow for various testing, component, and repeatability uncertainties. Additional margin to allow for degraded voltage, current, and temperature conditions which may be present during a design basis accident are also factored into the evaluation. Generic direct current (DC) motor performance curves supplied by the vendor are used to support determination of available motor torque and thrust and valve operating speed. When using the generic performance curves, a +/- ten percent error margin is applied to the calculation to account for potential performance differences between the generic DC motor curves and the specific DC motor installed in the MOV being evaluated.

The licensee recently sent four spare DC motors to an off-site test facility to develop specific performance curves unique to each individual motor. The inspector reviewed procurement documents and verified that the test facility was on the licensee approved vendor list for safety-related services. The licensee properly documented certification of the test facility to the quality standards of 10 CFR 50, Appendix B. Each of the four motors demonstrated performance superior to that of the generic vendor performance curve. Available motor torque and speed were greater than those of the generic curve for the anticipated range of operating electrical current.

To improve MOV operability margin, the DC motor for the high pressure coolant injection system outboard steam supply valve (2301-5) actuator was replaced with one of the tested spare motors during a recent plant outage. The motor specific test data supported elimination of the minus ten percent generic motor curve margin. The licensee reevaluated the available thrust and torque and validated the improved operability performance margin. The available closure thrust at design conditions improved to 43,979 pounds. This was greater than the required closure thrust of 32,415 pounds and less than the manufacturer's nominal operator thrust rating of 45,000 pounds. The inspector independently reviewed the motor specific test results and engineering calculation M-565 (revision 4), "MOV Thrust Calculation for MO-2301-5." Test data, valve characteristics, and assumptions were properly developed. The engineering evaluation was detailed, well documented, and supported the operability determination. The inspector noted that valve and actuator replacement for 2301-5 are planned for implementation during the next refueling outage to restore full design margin. Replacement of the actuator DC motor to improve current operability margin pending valve replacement was a positive initiative.

4.2 Motor Operated Valve Design Restoration Status Meeting

The motor operated valve (MOV) project team has developed a detailed plan to accomplish maintenance, testing, and modifications intended to restore full design margin to ninety safety related MOVs in accordance with NRC Generic Letter 89-10. The current schedule implements the plan over the next two refueling outages. The inspector observed a recent MOV project meeting which was attended by approximately twenty people representing various disciplines within the licensee organization. Detailed discussions were conducted ranging from the status of design requirement validation, dynamic performance testing, proposed modifications, and impact of recent industry experience, to resources, training, and scheduling of planned maintenance. The inspector specifically noted close coordination between the design engineers which developed associated design changes and the outage scheduling group. Maintenance to resolve generic issues, design changes, and performance testing of individual valves were effectively integrated on the outage schedule. The greatest challenge identified by the team was the need to properly train work crew supervisors to support the large number of maintenance and testing tasks that are scheduled to occur in parallel to meet the outage deadline. Meetings of this nature are conducted monthly to ensure the program remains on track. The inspector concluded that appropriate resources had been assigned and that the MOV project team was effectively communicating the MOV betterment plan throughout the organization.

4.3 Intermittent Operation of High Pressure Coolant Injection Suction Valve

On February 7, 1994, the high pressure coolant injection (HPCI) suction valve (2301-6) failed to fully close during a periodic stroke timing test. No cause was readily apparent. The valve was then cycled successfully five times with good cycle time repeatability and no indication of performance degradation. The licensee initiated problem report 94.9052 and performed troubleshooting on 2301-6 during the February 22, 1994 MSIV forced outage when the HPCI system was not required to be operable. A root cause analysis team (RCAT) was established and a detailed test plan developed to diagnose the failure. Technicians performed a visual inspection of the valve and measured various diagnostic parameters during valve stroking in attempts to repeat and localize the failure.

A voltage spike, occurring 0.3 seconds after valve actuation in the closed direction, was identified and localized to the torque switch. The torque switch was replaced and the spring pack was removed from the actuator for inspection. Further evaluation attributed the voltage spike to a hammerblow impact which occurs upon initial actuator movement when the worm gear engages the drive sleeve. By design the drive sleeve allows the worm gear to rotate without stem load as clearances in the gear train are taken up until contact with the drive sleeve lug occurs. Upon engagement, the force on the drive train creates a hammerblow effect which may cause the torque switch to open. If the torque switch is not by-passed in the circuit at the time of the hammerblow, and the force exceeds that of the torque switch setting, power to the motor is interrupted. Technicians also identified an undesired gap between internal components of the torque switch spring pack which unbalanced the torque switch. This contributed to the magnitude of the voltage spike caused when the torque switch opened. Technicians eliminated the spring pack gap and properly reinstalled the spring pack.

The RCAT concluded that the failure of 2301-6 to close on February 7, was due to hammerblow occurring after the closed torque switch bypass dropped out. Fast acting electrical contactors for this valve were installed during the previous refueling outage to address a previously identified potential overthrust concern. An unanticipated consequence of installing the fast contactors was the fact that the worm gear hammerblow may now occur while the torque switch function is in the circuit. Previously, with slower contacts, the torque switch had been bypassed when the initial hammerblow occurred. The RCAT recommended replacing the existing two pole limit switch with a four pole limit switch assembly. This modification provides greater flexibility which permits the torque switch to be bypassed for a greater portion of the valve travel in closure direction. Plant design change (PDC) 94-15 subsequently installed a four pole limit switch assembly set to bypass the torque switch in the manual closing logic until the valve is approximately 15 percent closed. This change maintains the torque switch bypassed until valve starting transients dissipate. For automatic close actuation, the torque switch remained bypassed for approximately 98 percent of the closure stroke. The inspector reviewed the safety evaluation for PDC 94-15 and concluded that the modification improved the reliability of 2301-6 and did not constitute an unreviewed safety question. Appropriate post maintenance testing was successfully completed.

In the automatic mode of operation, the torque switch is bypassed for 98 percent of valve travel in both the open and close directions. Therefore the problem described above did not at any time affect the safety-related operability requirements for the 2301-6 valve. Sustained effort by the RCAT resulted in the identification and correction of an elusive problem. The RCAT identified 40 additional safety-related valves which had a two pole rotor limit switch assembly design and reviewed existing diagnostic test data to determine if they were susceptible to the same problem. Eighteen valves have sufficient torque switch bypass duration. Two valves are subject to the starting transient with the torque switch in the circuit. However, in each case, the magnitude of the starting transient is insufficient to open the torque switch. The remaining twenty valves have not undergone diagnostic testing which can be used to identify this phenomenon. Review criteria have been revised for future diagnostic testing to identify those valves in which the hammerblow transient occurs with the torque switch in the circuit. In addition the licensee is evaluating replacement of all remaining two pole rotor limit switch assemblies in safety-related valves with four pole assemblies. The inspector concluded that licensee troubleshooting and evaluation of the 2301-6 valve failure to close was detailed and far reaching. Licensee effort to definitively diagnose and correct the problem generically instead of focusing solely on the 2301-6 valve demonstrated a strong safety perspective.

4.4 Failure of Motor Operated Valve SMB-1 Torque Switch Roll Pin

On March 9, with the reactor at approximately 100% of rated power, the high pressure coolant injection (HPCI) system was declared inoperable when a motor operated valve (MOV) failed in the closed position during valve operability surveillance testing. A motor overload alarm was received and the power supply circuit breaker tripped when the valve (2301-9) was stroked closed. Operators verified that the valve indicated closed, found the electrical supply breaker tripped, and noted that the valve actuator motor was hot to the touch. The 2301-9 valve is a normally open test and maintenance valve on the pump discharge line, that receives an open signal upon initiation of the HPCI automatic start logic. The valve is operated by a Limitorque SMB-1 actuator. On March 9, the valve was closed to maintain the pressure isolation boundary during inservice stroke time testing of the normally closed HPCI system injection valve.

Troubleshooting determined the SMB-1 torque switch roll pin had sheared and prevented the torque switch from opening and deenergizing the supply power to the actuator as designed when the valve reached the full valve closed position. A subsequent engineering evaluation concluded that the integrity of the valve disk and stem was maintained. The valve was opened manually and clamped in the open position using temporary modification (TM) 94-05. The engineering evaluation and TM 94-05 were detailed and technically sound. The inspector independently verified that the stem clamp was properly installed. Following installation of TM 94-05 the periodic HPCI pump operability surveillance was successfully performed to verify that the system flow path was not obstructed. The HPCI system was returned to normal standby service and declared operable on March 10.

The actuator was removed from the valve, disassembled, and inspected for signs of damage. Problem report 94.9104 was initiated to determine root cause of the 2301-9 valve failure and develop a comprehensive corrective action plan to preclude recurrence in any of the safety related MOVs. Metallurgical analysis identified the roll pin failure mechanism as brittle fracture from a single overload or impact event. The torque switch shaft was determined to be made of appropriate material for this application. Engineers concluded that the roll pin failed due to a material or design deficiency. The roll pin material is AISI 1070 high carbon steel which is brittle and hard (Rockwell "C" 47.5). The original design may have underestimated the magnitude of the torsional forces placed upon the torque switch shaft and pinion gear roll pin during valve starting and stopping transients. The licensee promptly notified the industry of the event via the nuclear network.

The licensee identified several factors which contributed to the magnitude of the valve starting forces. (1) A large unseating pullout torque was generated when the flex wedge disk was freed (sticking disk) from it's closed seat. This resulted in a large starting transient impact (hammer blow) on the pinion. The hammer blow effect of the gate freeing its seat is distinct from the designed worm gear to drive sleeve hammer blow, and creates excessive torsional forces that have the potential to shear the torque switch roll pin. The licensee concluded that this hammering effect was not considered in the actuator design. Additionally, it appears that the failure mode would occur on an opening stroke only and that gate valves are most suspect to the failure. (2) An older style spring pack was installed with no grease relief mechanism. Hard grease found inside the spring pack Belville washers, restricted the spring pack compression. (3) Higher torque switch settings can result in greater spring pack displacement. The 2301-9 valve had a higher torque switch setting than similar valves. The problem report was thorough and comprehensively evaluated safety related valves throughout the reactor plant for susceptibility.

The inspector observed nondestructive examination of the valve actuator and actuator housing. In addition to the failed torque switch roll pin, technicians identified several raised bolting holes and minor circumferential cracks around one of the eight bolt holes on the actuator upper thrust bearing housing. The crack was not initially visible to the eye, but was identified using magnetic particle testing. Engineering determined that the crack did not impact valve operability. However, the housing fatigue limit was exceeded. Additional engineering evaluation determined that the actuator rated thrust and torque values had also been exceeded. Short term corrective actions included refurbishment of the actuator with new internal components and replacement of the housing for the 2301-9 actuator.

Previously, the vendor had issued 10 CFR Part 21 notifications to licensees in 1990 and 1991 regarding roll pin failures on SMB-00 torque switches. In SMB-00 applications, spring pack movement is transmitted to the torque switch electrical contacts via a lever arm and actuation link. In SMB-0 through SMB-5 applications, spring pack movement is transmitted to the torque switch electrical contacts via a rack and shaft mounted pinion. The licensee industry experience review program recently identified multiple similar failures of the roll pins on the rack and pinion model torque switch used in SMB-0 through SMB-5 Limitorque actuators at several other facilities.

The licensee conducted a detailed evaluation to determine if further 10 CFR Part 21 applicability exists. Forty-seven additional safety related MOVs which use the same 3/32 inch torque switch roll pin were evaluated using the following considerations. Roll pin failure causes the torque switch contacts to remain closed by the torque switch springs, even when a high torque condition exists. This is not a credible concern on the valve opening stroke, because the valve either stops on limit or the torque switch is backed up by a limit switch. However, in the closing direction with a failed pin, the torque switch contacts will not trip the motor. If the valve is not stopped on limit, the valve will fully close into the seat at stall torque and possibly burn out the actuator motor. The engineering evaluation assumed that the valve would be unavailable to function further following closure at stall torque. The evaluation also noted that for most failure scenarios, roll pin failure would be identified during routine surveillance testing. The licensee concluded that the existing potential for roll pin failure would not prevent the specified valves from performing their safety function and therefore did not constitute a substantial safety hazard at Pilgrim station. The licensee closely communicated with the vendor and shared information from their causal analysis. Following detailed discussions with the licensee, the vendor issued a Potential 10 CFR Part 21 Condition notification to the industry on March 23, 1994. The notification did not fully document the root cause of the roll pin failure, but did recommend replacement of existing 3/32 inch roll pins with 1/8 inch roll pins which are made of a more appropriate material.

Long term corrective actions identified through PR 93.9104 were comprehensive. The MOV project team subsequently integrated vendor recommendations, to upgrade the torque switch roll pin and to replace original spring packs with improved internal grease relief spring packs, into the existing MOV betterment schedule. The 2301-9 valve will be disassembled and inspected in an upcoming outage. Action items to update the warehouse inventory of spare spring packs and torque switches have been assigned. Corrective actions were assigned with appropriate priorities. Licensee efforts to establish the root cause for the 2301-9 failure and communicate this information to the industry were noteworthy.

5.0 PLANT SUPPORT (71707)

5.1 Packaged Bottle of Wine Within the Protected Area

On Monday March 14, 1994, a packaged and unopened bottle of wine was delivered to the office of a division manager located within the protected area. Upon discovery of the wine, the manager immediately notified security section supervision who assumed possession of the package. A security investigation determined the packaged wine was a nominal gift from the food service vendor for the new support building cafeteria that is located outside of the protected area. The investigation concluded the package had been delivered by a private shipping company and had been received within a single larger shipment at the BECo warehouse on

Friday March 11, 1994. This conclusion was corroborated by security computer records, and interviews with warehouse personnel, shipping company distribution center representatives, and security force members. The involved security patrolmen responsible for the search/x-ray officer function and the observing officer function stated that the entire shipment had been passed through the x-ray machine at the warehouse receiving area.

Security supervision verified the x-ray machine was functioning properly and also passed the package of wine through the machine and determined it was readily identifiable on the monitor. The licensee concluded the search/x-ray officer performed an inadequate search of the shipment, in that either the packages were improperly loaded onto the x-ray machine which may have caused the wine bottle to be obscured or the officer did not properly observe the monitor during the search. The security patrolman has been retrained on search procedures and was counselled on attention to detail. The licensee properly recorded this event as a loggable event in accordance with 10 CFR 73.71.

The inspector independently reviewed the applicable security procedures, toured the warehouse security control point, observed a test block demonstration of x-ray functionality, discussed the event with security section management, and reviewed the event investigation report. The inspector concluded that the licensee response to this event was appropriate and had no further questions.

5.2 Soil Sampling

The inspector reviewed the gamma spectrometry analytical results from six soil samples which were obtained from various locations around the satellite parking area of the new engineering support building. The results indicated that only naturally occurring radioactive materials were present in the soil samples. A review of sample descriptions indicated the six samples were representative of the overall content of the soil present in the parking area and adjacent to the new support building. The samples consisted of top soil or subsoil or a mixture of both types of soil.

The samples were analyzed using a Marinelli Beaker counting geometry and were counted to a lower limit of detection (LLD) which exceeded the requirements in the Technical Specifications/Offsite Dose Calculation Manual (ODCM) for radiological environmental monitoring. The inspector had no questions regarding this sampling methodology.

6.0 SAFETY ASSESSMENT AND QUALITY VERIFICATION (40500)

6.1 Licensee Event Report Review

The inspector reviewed Licensee Event Reports (LERs) submitted to the NRC to verify accuracy, description of cause, previous similar occurrences, and effectiveness of corrective actions. The inspectors considered the need for further information, possible generic implications, and whether the events warranted further onsite follow-up. The LERs were also reviewed with respect to the requirements of 10 CFR 50.73 and the guidance provided in NUREG 1022 and its supplements.

- **LER 93-25**

LER 93-25, "Reactor Core Isolation Cooling (RCIC) System Made Inoperable Due to Minimum Flow Valve Position During Surveillance Testing," dated November 22, 1993, describes the October 24, 1993 failure of the RCIC minimum flow valve (1301-60) to close. Troubleshooting identified the cause to be a loose connector and a degraded auxiliary relay within the electrical supply circuit breaker. Irregular turbine control valve indication was observed and evaluated as documented in NRC Inspection Report No. 50-293/93-20. An unplanned RCIC system isolation resulted from incomplete venting of the hydraulic control oil system following replacement of the turbine control valve servo mechanism. Corrective actions were promptly identified and implemented. Operability surveillances were successfully completed and the RCIC system was declared operable on October 25, 1993. Alternate emergency injection systems including the high pressure coolant injection system were operable at the time of this event. The LER properly addressed all reporting criteria.

- **LER 93-26**

LER 93-26, "Low Reactor Vessel Water Level While Shutdown Resulting in Automatic Scram Signal and Containment System Isolations", dated December 6, 1993, describes the November 8, 1993 automatic reactor protective system (RPS) actuation which occurred when operators failed to recognize and respond to diverging reactor vessel (RV) water level indication during preparations for reactor restart on November 8, 1993. The reactor was subcritical and all control rods were fully inserted prior to the RPS signal. This event was of minimal safety significance due to plant conditions and availability of safety systems at the time of the event. The root cause, plant response, and corrective actions are fully documented in NRC Inspection Report No 50-293/93-20. The LER provided a complete description of the event and thoroughly developed the causal analysis. The LER properly addressed all reporting criteria.

The licensee determined that the root cause was operator error, with the presence of noncondensible gasses in the instrumentation sensing lines classified as a contributing cause. However, as previously documented, the inspector concluded that the root cause of the event was the existing design of the sensing lines for the feedwater system RV level instruments. The licensee has initiated an engineering service request to review the option of installing a design change on the feedwater instrument sensing lines similar to the backfill system currently installed for the safety related RV level instruments (NRC Bulletin 93-03). This action was appropriate to address the issue of root cause resolution.

7.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (30702)

7.1 Routine Meetings

At periodic intervals during this inspection, meetings were held with senior BECo plant management to discuss licensee activities and areas of concern to the inspectors. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting on April 19, 1994, summarizing the preliminary findings of this inspection. No proprietary information was identified as being included in the report.

7.2 Management Meetings

On March 17, 1994, NRC Licensee Meeting Number 94-41 was convened in the NRC Region I office to discuss the BECo response to the December 13, 1993 storm that presented significant challenge to plant operators and intake structure systems. This event and the associated NRC concerns are documented in NRC Inspection Report 50-293/93-23. The overhead slides used by BECo during the meeting are included as Attachment I to this report.

On March 4, 1994, Mr. Ed Kraft, Vice President of Nuclear Operations and Station Director (VPN & SD) announced his resignation from BECo effective on March 31, 1994, to accept a Senior Vice President position at the Commonwealth Edison Quad Cities facility located within NRC Region III. Mr. E. Thomas Boulette, Senior Vice President, Nuclear, announced that he would assume VPN & SD responsibilities until a permanent replacement is named.

7.3 Other NRC Activities

On March 7-11, 1994, a Region I Senior Reactor Engineer conducted the first week of a planned two week initiative inspection of safety relief valve performance and monitoring programs. However, the inspector sustained an injury in a non-work related accident that will delay the completion of the inspection until a later date. No significant issues were identified during the initial inspection effort.

During this inspection report period, Mr. Eugene M. Kelly, Chief of Reactor Projects Section 3A, with responsibility for the management of NRC activities at Pilgrim was assigned as Chief of the Plant Systems Section within the Division of Reactor Safety (DRS). Mr. Richard J. Conte, Chief of the Boiling Water Reactor Section within DRS was assigned to succeed Mr. Kelly.

ATTACHMENT

BECO SLIDE PRESENTATION

NRC MANAGEMENT MEETING

MARCH 17, 1994

BOSTON EDISON COMPANY
PILGRIM NUCLEAR POWER STATION

MANAGEMENT MEETING TO DISCUSS DEC. 13, 1993 STORM

Boston Edison

March 17, 1994

TEAM PRESENTATION

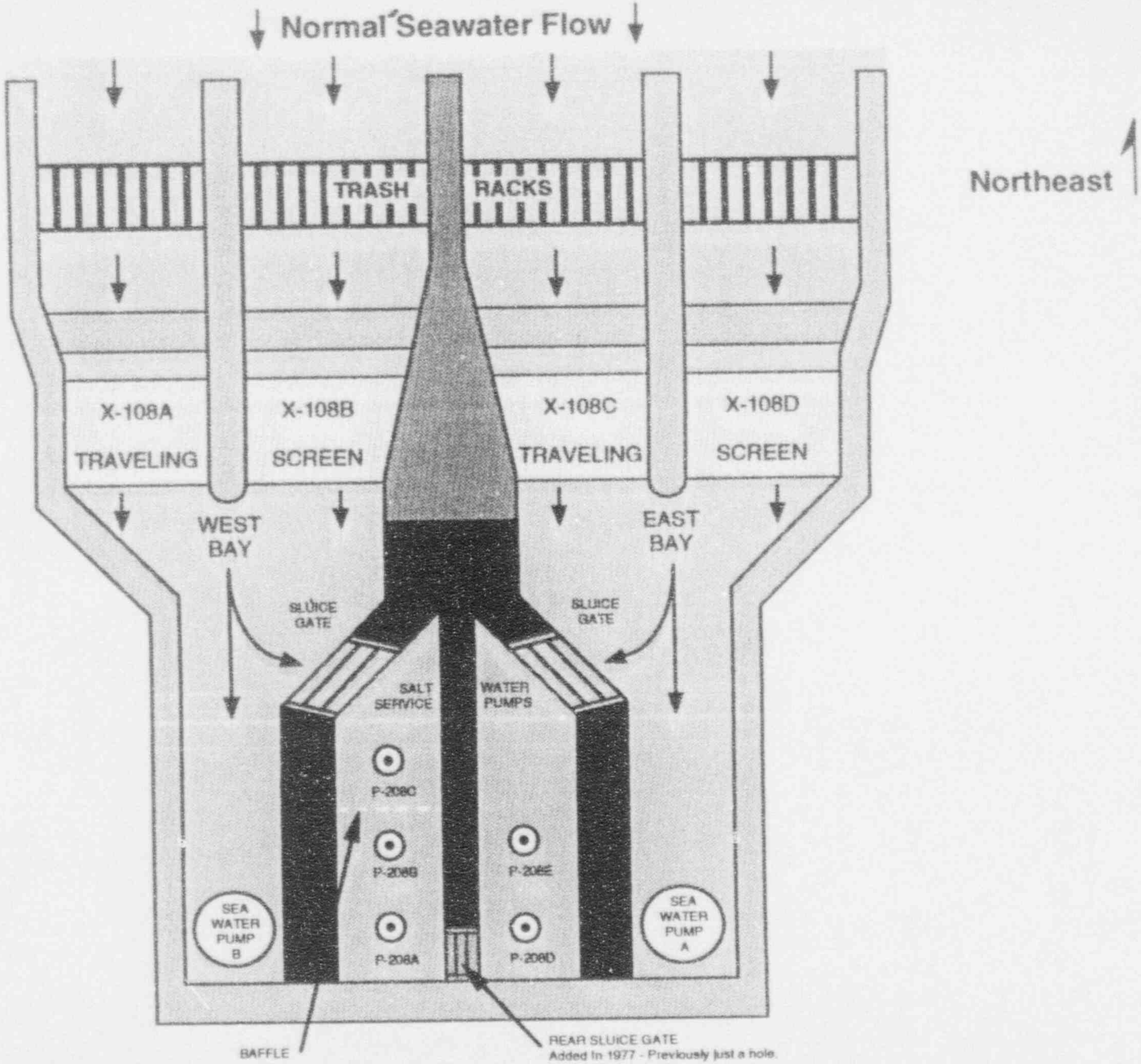
- Introduction/Meeting Overview — E T B
  - Senior Management Response
  - Management Expectations
- December 13 Storm & Screenhouse Performance — L L S / T A S
  - System Configuration
  - Storm Preparation
  - Response to Events
  - Follow-up Actions
- Technical Overview — W C R
  - Root Cause
  - Team Assessment Results
  - Long Term Corrective Actions
- Summary / Closing — E T B

SENIOR MANAGEMENT RESPONSE DIRECTED TO SAFETY

- Management Expectations are Focused on Nuclear Safety
- Timely and Comprehensive Event Response is Expected
- Lessons Learned from the December 13 Storm have Strengthened Performance

DECEMBER 13, STORM/ SERVICE WATER SYSTEM PERFORMANCE

- Intake Structure and System Description
- Preparations Taken Prior to the Storm
- Operations Command and Control
- Operations Actions after Event



INTAKE STRUCTURE 21'-6" LEVEL
PARTIAL PLAN VIEW

STORM PREPARATIONS

- Traveling Screens/Screen Wash Pumps in Service
- Operator Stationed in Screenhouse
- System Engineer Monitoring Screenhouse
- Normal Configuration of Sluice Gates
- Operating One SSW Pump in each Train

CONDITIONS DETERIORATE DUE TO WEATHER

- Second Operator in Screenhouse
- Seawater Bay Levels Oscillating Due to the Wave and Storm Effects.
- Heavy Seaweed Loading of the 'C' and 'D' Screens
- Traveling Screen Shear Pins Break

EFFECTIVE OPERATIONS RESPONSE

- Reactor Power Reduced
- Seawater Pump 'A' Manually Tripped
- Rear Sluice Gate Opened
- East Seawater Bay Level Increased to Normal
- Broken Shear Pins Replaced
- Traveling Screens Returned to Service

IMMEDIATE ACTION TAKEN

- Briefed all Shift Crews Regarding the Event
- Standing Order Issued for Rear Sluice Gate In Open Position
- Use of EPIC Computer to Monitor/Trend Seawater Bay Level

ACCELERATED SCREENHOUSE IMPROVEMENT

- Screenwash Motors
- Screenwash Pumps
- New Screenwash Header (fresh water)
- New Screenwash Strainers
- Foam Suppression Header (fresh water)

EVENT REVIEW TEAM ACTIONS PRECLUDE RECURRENCE

- Procedures
- Formal Training
- Additional Hardware Improvements

ADDITIONAL ACTIONS IMPROVE OVERSIGHT

- Revised Critique Procedure
- Revised Format 0800 Meeting
- More Frequent PAC Attendance
- More Frequent ORC Attendance
- More Time in the Control Room

TECHNICAL OVERVIEW

- Root Cause Analysis
- ERT Assessment
- Short-term Corrective Actions
- Long-term Corrective Actions

ROOT CAUSE ANALYSIS

- Heavy Influx of Seaweed During Storm
  - Traveling Screen Shear Pin Failure

- Contributing Causes
  - 'C' & 'D' Traveling Screen Speed
  - 'A' Seawater Pump Operation
  - Annunciator Response Procedure
  - Training on Shear Pin Replacement
  - Intake Structure Level Alarms/ Level Monitoring

ERT ASSESSMENT BROADLY FOCUSED

- Review / Document Design Basis for Rear Sluice Gate
- Assure Instrument Adequacy
- Develop Storm Procedure
- Evaluate Shear Pins
- Provide Enhanced Operator Training
- Evaluate Screen Differential Alarm
- Evaluate Rear Sluice Gate Condition
- Evaluate SSW Pump Restraints
- Evaluate Intake Canal/Need for Dredging

Boston Edison

SHORT TERM ACTIONS PRECLUDE RECURRENCE

- Confirmed Rear Sluice Gate Design Basis - Design Met
- Procedures Revised, New Procedures Issued
- Traveling Screen Shear Pins Staged
- Operator Training Initiated - Ongoing
- Confirmed Screen Differential Alarm Setting Appropriate - Procedure Clarified
- Rear Gate Inspection - Satisfactory
- Seismic Adequacy Assessed - Satisfactory

LONGER TERM ACTIONS IMPROVE SERVICE WATER RELIABILITY

- Replace 'A' & 'B' Screens
- Upgrade Screenwash System
- Supplement Fresh Water Screenwash
- Instrumentation Enhancements
- Foam Suppression for Switchyard Reliability

PREVIOUS RELIABILITY IMPROVEMENTS

- Replacement of 'C' & 'D' Screens
- Replacement of SSW Piping
- Upgraded Hypochlorination System
- Planned Intake Canal Dredging

SUMMARY AND CONCLUSIONS

- Nuclear Safety is Foremost in Our Minds
- We have Taken Full Advantage of the Lessons Learned from the Event
- We have Communicated Our Expectations
- We are Committed to Personal Follow-up and Management Oversight