U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 50-293/88-25

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

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: June 1 - July 17, 1988

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Inspection Summary:

Areas Inspected: Routine resident inspection of plant operations, radiation protection, physical security, plant events, maintenance, surveillance, outage activities, and reports to the NRC. Principal licensee management representatives contacted are listed in Attachment I to this report.

Results:

Unresolved Items:

1. The licensee identified significant cracking in the yokes of two residual heat removal system motor operated valves. The licensee's root cause analysis and corrective actions for the valve yoke failures will be reviewed in a future inspection (Section 3.c, UNR 88-25-01).

2. During the past seven months the licensee has experienced about 19 engineered safety feature (ESF) actuations. In several cases, similar actuations had previously occurred. The licensee's effort to identify the root causes, and to reduce the number of ESF actuations will be reviewed during a future inspection (Section 4.d, UNR 88-25-02).

Strengths:

1. The licensee has established and implemented a strengthened maintenance control process. Observation of its implementation indicates that it provides significantly improved guidance and control of field activities (Section 3.a).

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DETAILS

1.0 Summary of Facility Activities

The plant has been shutdown for maintenance and to make program improvements since April 12, 1986. The reactor core was completely defueled on February 13, 1987 to facilitate extensive maintenance and modification of plant equipment. The licensee completed fuel reload on October 14, 1987. Reinstallation of the reactor vessel internal components and the vessel head was followed by completion of the reactor vessel hydrostatic test. The primary containment integrated leak rate test was also completed during the week of December 21, 1987. During this period, the licensee performed routine maintenance and surveillance tests. Some of the ongoing projects include: degraded voltage protection modifications, DC electrical breaker testing, and RHR valve repair work. On June 21, the licensee announced that effective August 17, 1988 the contract security services for Pilgrim will be provided by Wackenhut Security. Presently, these services are provided by Globe Security.

NRC inspection activities during the report period included: 1) a review of the licensee's corrective actions on previous NRC inspection findings during the week of June 6, 1988, 2) a review of the bases for licensee conclusions documented in their Self-Assessment Report during the week of June 20, 1988, and 3) a review of the licensee's corrective actions on previous NRC electrical inspection findings during the weeks of June 20 and June 27, 1988.

On June 2, 1988, Mr. James M. Taylor, NRC Deputy Executive Director for Regional Operations, and Mr. William T. Russell, Regional Administrator, Region I were onsite and toured the station. A public briefing of the NRC Commissioners was conducted on June 9, 1988 at the NRC offices in Rockville, Maryland regarding the status and progress of the Pilgrim plant. During the week of July 4, 1988, a Systematic Assessment of Licensee Performance (SALP) Board meeting was held for Pilgrim at the NRC Regional office in King of Prussia, Pennsylvania. The SALP assessment covered the period of February 1, 1987 through May 15, 1988. On July 12, 1988, Dr. Eric S. Beckjord, Director, Office of Nuclear Regulatory Research, and Dr. Denwood F. Ross, Deputy Director, Office of Nuclear Regulatory Research were onsite and toured the station.

The NRC announced that a comprehensive Integrated Assessment Team Inspection (IATI) has been scheduled to begin on August 8, 1988. Onsite preparation for the IATI will be conducted during July 19-21, 1988.

2.0 Followup on Previous Inspection Findings

Violations

(Closed) Violation (87-50-05), Failure to Notify the NRC as Required by 10 CFR 50.72. On two occasions the licensee experienced automatic actuations of the primary containment isolation system, an engineered safety feature (ESF), resulting in the isolation of the reactor water cleanup system. These ESF actuations were not reported to the NRC within the required four hour time period. In response to the Notice of Violation, the licensee committed to prepare and proceduralize a list of Pilgrim Specific ESF's, and to discuss the 10 CFR 50.72 reporting requirements during the next annual requalification program for licensed operators.

The licensee has revised the applicable procedure to clarify guidance on reportability of safety system actuations. The licensee developed a list of Pilgrim Engineered Safety Features and incorporated this list into Procedure No. 1.3.24, "Failure and Malfunction Reports". Additionally, a memorandum was issued to senior shift operations personnel providing clarification of the 10 CFR 50.72 requirements for reporting ESF actuations. The licensee is providing training during the ongoing annual requalification program for licensed operators on reporting requirements; the reportability of ESF actuations is specifically discussed. No recent deficiencies have been noted in this area. It appears that operations personnel have become familiar with these reporting requirements through the clarifications provided by the memo and appropriate procedure changes. Based on the above, this item is closed.

Unresolved Items

(Closed) Unresolved Item (85-20-03), Removal of Safety-Related Equipment from Service. This item was last updated in inspection report 50-293/85-26. During inspection 85-20 the inspector expressed concern that the licensee had not taken action to prevent reactor protection system (RPS) and primary containment isolation system (PCIS) instrumentation from being removed from service for extended periods of time during surveillance testing. This problem was identified on August 12, 1985 when main steam line high radiation monitors were removed from service. The "B" monitor was left partially disconnected and unable to trip, leaving the RPS and PCIS systems with less than the minimum number of operable instrument channels per trip system required by Technical Specifications. At the time, there existed no formal time limit for keeping safety-related RPS and PCIS equipment out of service during testing and calibration.

In response, the Plant Manager issued a memo to operations and maintenance personnel restricting the time that an RPS or PCIS instrument may be taken out of service for surveillance testing without tripping the channel to two hours. This policy was consistent with the existing BWR Standard Technical Specifications. Subsequently, a change was issued to the Pilgrim Technical Specifications restricting the time that an RPS or PCIS instrument channel may be taken out of service for surveillance testing; the limit is six hours. NRC acceptance of this six hour time period was based on an NRC safety evaluation of the BWR Owners Group generic analyses, and the licensee's demonstration of the applicability of these analyses to Pilgrim. Based on the issuance of the Technical Specification change, this item is closed.

(Closed) Unresolved Item (87-34-01), Reactor Building Auxiliary Bay and Intake Structure Seismic Qualification. This item was reviewed in detail during inspection 50-293/87-45. During that inspection, the licensee demonstrated through documentation the seismic adequacy of the Reactor Building Auxiliary Bay to house safety-related components. The licensee also demonstrated the seismic adequacy of the portion of the Intake Structure Housing the Salt Service Water Pumps, but found that the remaining portions of the Intake Structure could not be seismically qualified. The licensee identified safety-related conduit carrying Salt Service Water System cables that were routed outside the safety-related (Class I) portion of the Intake Structure. As a result, the licensee appropriately informed the NRC via the ENS of this finding and issued Licensee Event Report 87-009. In addition, Safety Evaluation No. 2225 was performed to evaluate the operability of the Salt Service Water System assuming the loss of these circuits. Based on this evaluation, it was concluded that the reactor could safely be maintained in the cold shutdown condition during refueling. The licensee then proceeded with their planned refueling operations. Subsequently, Plant Design Change (PDC) No. 87-64 was issued to reroute the affected safety-related conduit through the Class I portion of the Intake Structure.

The inspector reviewed LER 87-009, Safety Evaluation No. 2225 and PDC 87-64 and other documents and confirmed that the design changes detailed in PDC 87-64 had been completed. The inspector examined portions of completed PDC 87-64 modifications (i.e. the rerouting of conduits A154, X158 and B3690) and confirmed that the safety-related circuits had been rerouted through the Class I portion of the Intake Structure. No violations were noted. This item is closed.

(Closed) Unresolved Item (87-50-01), Seismic Qualification of Hydraulic Control Units (HCUs). The inspector reviewed the licensee's response to concerns posed in inspection report 50-293/87-50 about the adequacy of the supports for the hydraulic control units (HCUs). The licensee's response and Engineering Service Request (ESR) 88-146 revealed that all the hydraulic control units, including the HCU that is not tandem mounted, are adequately supported to ensure seismic qualification. The inspector reviewed the analysis in detail. No problems were noted. This item is closed.

<u>(Closed) Unresolved Item (88-12-01), Deficiencies in Procedure 8.9.8, "Battery Rated Load Discharge Test."</u> The inspector reviewed licensee procedure 8.9.8, Rev. 16, dated June 10, 1988. The procedure deficiencies identified in inspection report 50-293/88-12 have been corrected. In particular, the procedure now includes a correction factor to take into account the change in battery capacity with temperature, and requires the use of a calibrated hydrometer for measuring specific gravity. The inspector also reviewed the last completed version of procedure 8.9.8, as well as procedure 3.M.3-42, Rev. 1, "Battery Charger Maintenance and Calibration". Both of these procedures are performed once every refueling outage . No problems with the procedures were noted. This item is closed.

<u>Inspector Follow Items</u>

<u>(Closed) Inspector Follow Item (85-01-06), Error in Computer Program Used for Technical Specification Surveillance.</u> During inspection 50-293/85-06 a discrepancy was noted between the tested and reported leak rates during performance of drywell to torus vacuum breaker leak rate testing. The licensee determined that an error in a computer program had caused the leak rate to be reported incorrectly. Recalculations of several previous leak rate test results by the licensee showed another error.

The licensee conducted a survey in 1985 to determine the extent to which computer programs were used to ensure Technical Specification compliance. The results of the survey were updated during April and May, 1988. The survey consisted of identifying computer programs used, and the method used to validate program accuracy. The accuracy of each computer programs was revalidated. Procedure 1.3.46, "Documentation and Qualification Procedure for Safety-Related Computer Code and Data Bases" is now used to ensure continued accuracy. This procedure establishes the requirements for documentation and qualification of computer programs used for safety-related analyses, including how revisions to the program are to be handled. Based on the above this item is closed.

<u>(Closed) Inspector Follow Item (85-26-03), Provide Specific Guidance on the Use of Maintenance Summary and Control (MSC) Attachments to Maintenance Requests.</u> This item was last updated in inspection report 50-293/87-50. The licensee formerly used the MSC attachment as a mechanism for describing and planning maintenance activities associated with approved maintenance requests (MR). Due to ongoing concerns in the area of control and documentation of maintenance tasks, the licensee recently implemented extensive changes to the MR process. As part of these changes the use of the MSC form was discontinued. The new MR process is discussed further in section 3.a of this report. This item is closed.

<u>(Closed) Inspector Follow Item (86-14-04), Review the Adequacy of the Residual Heat Removal System Minimum Flow Protection.</u> On May 19, 1986 the licensee reported that for the installed design at Pilgrim the failure of a single instrument could lead to loss of all four residual heat removal (RHR) pumps. At Pilgrim there are two RHR loops with two pumps per loop. Each pair of pumps shares a common minimum flow line. The licensee identified that failure of either pump minimum flow sensor could result in loss of all minimum flow paths, leading quickly to pump damage if no other discharge path is available. This situation could occur during a loss of coolant accident with reactor pressure remaining above 400 psig. IE Compliance Bulletin Number 86-01 was issued on May 23, 1986 addressing this problem. In responding to Bulletin 86-01 the licensee implemented Plant Design Change (PDC) 86-33. This modification changed the normal minimum flow valve position from closed to open, and removed the automatic valve closure signal, thereby ensuring a continuous flow path. Inspector review of licensee actions in response to Bulletin 86-01 and evaluation of PDC 86-33 were documented in inspection report 50-293/87-50.

During inspection 50-293/87-50, the inspector questioned the acceptability of a common minimum flow line for two RHR pumps. NRC Information Notice 87-59 described a case in which one of the two pumps sharing a common minimum flow line could be subjected to less than the required flow, due to unlike system operating characteristics. This item was left open pending licensee evaluation of this potential problem. Subsequently, the NRC issued Bulletin 88-04, "Potential Safety-Related Pump Loss" further discussing this issue. During the current period the inspector reviewed the licensee's response to Bulletin 88-04, the revised system operating procedures, licensee calculations and RHR pump test data. The licensee has removed the restricting orifices from the RHR minimum flow lines thereby increasing the flow rate from 300 gpm to about 500 gpm. The adequacy of 500 gpm minimum flow per pump was demonstrated by licensee calculation M-517-2, and by observation of pump internal condition during recent disassembly. The adequacy of a common minimum flow line for two pumps was demonstrated by licensee calculation M-517-1. Because the common section of piping is 3 inches, while piping associated with the individual pumps is 2 inches, the head loss contribution from the common piping is relatively small. Based on the calculations it appears that variations in pump performance within the allowable pump operating range would not adversely affect the minimum flow protection. Recent pump performance data reviewed by the inspector supports the accuracy of the licensee calculation. The licensee has also revised system operating procedures to minimize the duration of pump operation at minimum flow.

The inspector also reviewed the licensee's response to NRC Bulletin 88-04 dated July 13, 1988, relative to other safety-related systems. The information contained in the response appears accurate and does not indicate any additional areas of concern. Based on the above, this item is closed.

3.0 Routine Periodic Inspections

The inspectors routinely toured the facility during normal and backshift hours to assess general plant and equipment conditions, housekeeping, and adherence to fire protection, security and radiological control measures. Inspections were on weekends on June 4, 5, 11, 12, 18 and 26, 1988 for 33 hours and on July 2, 10 and 16, 1988 for 13 hours. Ongoing work activities were monitored to verify that they were being conducted in accordance with approved administrative and technical procedures, and that proper communications with the control room staff had been established. The inspector observed valve, instrument and electrical equipment lineups in the field to ensure that they were consistent with system operability requirements and operating procedures.

During tours of the control room the inspectors verified proper staffing, access control and operator attentiveness. Adherence to procedures and limiting conditions for operations were evaluated. The inspectors examined equipment lineup and operability, instrument traces and status of control room annunciators. Various control room logs and other available licensee documentation were reviewed.

The inspector observed and reviewed outage, maintenance and problem investigation activities to verify compliance with regulations, procedures, codes and standards. Involvement of QA/QC, safety tag use, personnel qualifications, fire protection precautions, retest requirements, and reportability were assessed.

The inspector observed tests to verify performance in accordance with approved procedures and LCO's, collection of valid test results, removal and restoration of equipment, and deficiency review and resolution.

Radiological controls were observed on a routine basis during the reporting period. Standard industry radiological work practices, conformance to radiological control procedures and 10 CFR Part 20 requirements were observed. Independent surveys of radiological boundaries and random surveys of nonradiological points throughout the facility were taken by the inspector.

Checks were made to determine whether security conditions met regulatory requirements, the physical security plan, and approved procedures. Those checks included security staffing, protected and vital area barriers, personnel identification, access control, badging, and compensatory measures when required.

a. Maintenance Process Restructuring

On May 27, 1988, Boston Edison management restricted performance of some maintenance tasks at Pilgrim. A licensee self-assessment completed on May 26, identified maintenance program weaknesses. Concerns regarding the effectiveness of the licensee's program for control of routine and corrective maintenance had been raised by the NRC during previous inspections, and during a recent maintenance team inspection completed on May 5, as documented in report 50-293/88-17. In addition, the licensee's Quality Assurance Department recently identified concerns in the area of maintenance on environmentally qualified (EQ) equipment. These QA concerns resulted in issuance of stop work order on all environmentally qualified equipment by the QA Manager. Licensee management concluded that changes to the station maintenance program were warranted. Maintenance tasks which were not covered by job specific, detailed, approved procedures were restricted. These actions were previously discussed in section 3.b of inspection report 50-293/88-19.

Early in the current inspection period the licensee completed the development of the revised maintenance control process. Several existing program procedures were extensively revised, and new procedures were created to provide specific methods and instructions for preparation and implementation of work plans and post-maintenance testing. Procedure 1.5.3, "Maintenance Requests (MR)", was revised to more clearly define the MR prioritization system, to incorporate previously implemented enhancements such as pre-job briefing instructions, to improve documentation and control of EQ maintenance and post-maintenance testing, and to provide for use of detailed job-specific work plans. Procedure 1.5.7, "Emergency Maintenance", was revised to provide more clear direction for processing of activities requiring rapid response by maintenance personnel. Procedure 3.M.1-30, "Post Work Testing Guidance", was also revised, and Section Instruction SI-MT-0501 was created, to strengthen control of post-work testing. The licensee created procedure 1.5.3.1, "Maintenance Work Plan" to detail the format, content and use of unique step-by-step instructions for each MR. Included in procedure 1.5.3.1 are enhanced provisions for control of special processes, documentation of EQ equipment maintenance and review of activities for potential plant impact. The licensee conducted training for maintenance, quality control and operations personnel on the new process prior to its implementation.

The inspector reviewed the revised licensee procedures discussed above, observed portions of the personnel training conducted, and evaluated application of the new process to several ongoing maintenance activities. The program appears to provide improved guidance to maintenance workers in the field, better control of abnormal configurations, and improved documentation of activities. Discussions with maintenance technicians and supervisors indicated that they were generally knowledgeable of, and comfortable with, the new process. The inspector reviewed several maintenance work plans in detail, including those associated with the disassembly and repair of failed Motor Operated Valve 1001-28B. The work plans appeared to contain sufficient detail to ensure proper performance and documentation of the activities. The inspectors will continue to monitor licensee efforts in this area.

b. Quality Assurance Stop Work Order on EQ Maintenance Activities

On May 19, 1988, the licensee's Quality Assurance (QA) Department issued a stop work order on certain maintenance activities involving environmentally qualified (EQ) equipment. A total of six QA deficiency reports (DR) had been issued by the licensee during the previous six months in the area of maintenance on EQ equipment. These previously issued DRs, combined with recent QA surveillance observations, prompted issuance of the stop work order. Weaknesses identified by the licensee included lack of control of the EQ Master List and EQ maintenance requirement documentation, poor replacement material traceability provisions, and weak training of maintenance personnel in EQ considerations and precautions.

During the current inspection period the licensee Quality Assurance Manager lifted the stop work order after completion of appropriate corrective actions by the licensee staff, and confirmatory review by QA auditors. Actions taken by the licensee included strengthening of the control and dissemination of EQ equipment maintenance requirements, personnel training, and implementation of the revised maintenance request process described in section 3.a above. While the stop work order has been lifted, many of the previously issued DRs remain open. The inspector will monitor ongoing licensee corrective actions, including any evaluation of the impact of these problems on the adequacy of maintenance tasks completed in the past.

c. Failure of Two Residual Heat Removal System Valve Yokes

On June 7, 1988, the licensee discovered cracking in a motor operated valve (MOV) yoke in the Residual Heat Removal System (RHR). The RHR system consists of two redundant loops with two pumps per loop. During operation of the Low Pressure Coolant Injection (LPCI) mode of the RHR system each loop injects to the reactor vessel through a single line at the reactor recirculation system. In addition to serving as LPCI injection paths the two lines serve as flow paths for shutdown cooling return. Each injection line contains a motor operated globe valve, a motor operated gate valve and a check valve in series. While attempting to remove the "B" loop of shutdown cooling from service the licensee was unable to secure flow when the globe valve, MOV 1001-28B, was closed. Followup inspections by the operations staff identified that the yoke had cracked about 270 degrees around at a weld between the lower yoke section and the motor actuator mounting plate. Subsequent inspection of the counterpart valve in the "A" RHR loop, MOV 1001-28A, identified indications of cracking in the lower portion of the yoke, just below the location of the crack in the MOV 1001-28B yoke. The licensee isolated the "B" RHR loop to facilitate disassembly of MOV 1001-28B. MOV 1001-28A was secured in the open position so that the "A" loop of shutdown cooling would remain functional. On July 10, 1988, the licensee made a formal notification of the failure to the NRC via ENS.

The licensee formed a task force composed of representatives of the nuclear engineering, operations, maintenance and system engineering departments. Reviews of the valve and motor operator design, and of the valve operating history were initiated. Plans were developed and implemented for removal of the MOV 1001-28B operator, yoke and valve internals. The damaged portion of the yoke was cut off and transported to the Massachusetts Institute of Technology (MIT) for materials analysis. The final MIT report was not issued by the close of the period. The motor operator was disassembled under the supervision of the cognizant system engineer and a representative of Limitorque, the operator supplier. No significant problems were identified. Inspection of the valve internals revealed cracking in the valve backseat stellite overlay. The damaged stellite was replaced. By the close of the inspection period the licensee had also removed the motor operator and yoke from the 1001-28A valve, and was about to commence disassembly and inspection of the operator.

The licensee is continuing to investigate the failures. The resident inspectors will continue to monitor licensee followup and findings. This item will remain unresolved pending licensee identification and correction of the failure root cause (UNR 88-25-01).

d. Licensee Actions in Response to the NRC Bulletin 88-05

NRC Bulletin 88-05, dated May 6, 1988 and Supplement 1 to the Bulletin, dated June 15, 1988 require that licensees submit information regarding materials supplied by Piping Supplies Incorporated (PSI), and West Jersey Manufacturing Company (WJM). It has been determined that these two companies have supplied potentially nonconforming piping materials to the nuclear industry. Licensees were requested to take actions to assure that any suspect materials comply with ASME Code and ASTM design and material specification requirements.

The licensee has identified a total of 186 flanges manufactured by either Piping Supplies Inc. (PSI) or West Jersey Manufacturing Company (WJM). Thus far, 18 flanges have been verified as installed in safety-related plant systems. Twenty-nine flanges from six different heat numbers in the warehouse were subjected to chemical and mechanical analysis. All chemical analyses were acceptable. Three flanges did not meet the tensile and yield strength specifications listed on the Certified Material Test Report (CMTR). The three flanges were identified as belonging to Heat No. CKS. The material was procured as ASTM SA-105 which has a required minimum tensile strength of 70 Ksi. The tensile strength on the CMTR was 81 Ksi. The measured tensile strengths of the three nonconforming components were 68 Ksi, 68 Ksi, and 69.5 Ksi. ASTM required minimum yield strength is 36 Ksi whereas the measured yield strengths were 35.5 Ksi, 37.2 Ksi, and 37.5 Ksi. The yield strengths reported on the CMTR was 59 Ksi.

The licensee is also conducting insitu-testing utilizing an Equotip hardness testing device. The measured hardness value is then equated to a Brinnell hardness number. After the close of the inspection, the licensee reported, on July 22, 1988, that an insitu-test of a 22-inch flange installed in the salt service water system indicated a Brinnell hardness number above the ASTM specifications. The Brinnell hardness number was 218 which is outside the 137-187 range for which no additional evaluation is needed. On July 22, 1988 the licensee notified the NRC Operations Center via a commercial line in accordance with the 48 hour reporting requirements specified in the IEB 88-05. Subsequent retesting of the flange resulted in acceptable hardness values. The licensee is continuing with their documentation review, inspection, testing, and evaluation. The licensee is also collaborating with the Nuclear Utility Management and Resources Committee (NUMARC) to share industry wide data in this area. The inspectors will continue to review licensee progress during future inspections.

e. Licensee Actions in Response to Bulletin 88-07

NRC Bulletin 88-07, dated June 15, 1988, was issued to ensure that adequate operating procedures and instrumentation are available, and that adequate operator training was provided, to prevent the occurrence of uncontrolled power oscillations during all modes of BWR operation. The Bulletin was issued in response to an event at the LaSalle Station during which the reactor experienced excessive neutron flux oscillations following a dual recirculation pump trip from power. The Bulletin required that all licensed reactor operators be made aware of the events at the LaSalle Station within 15 days of receipt of the Bulletin. In addition, within 60 days, the licensee is required to verify the adequacy of their procedures and operator training programs regarding actions to be taken in the event of uncontrolled power oscillations.

In response, the licensee distributed copies of Bulletin 88-07 and Information Notice 88-39 to all licensed operators. In addition, the licensee conducted a training session for licensed operators on the LaSalle event, as well as on operator actions required if uncontrolled power oscillations are experienced at Pilgrim. These actions appeared to meet the intent of the short-term requirements discussed in the Bulletin. Licensee engineering personnel are reviewing the adequacy of existing procedures in relation to this event. The inspectors will continue to monitor licensee followup to this Bulletin.

4.0 Review of Plant Events

The inspectors followed up on events occurring during the period to determine if licensee response was thorough and effective. Independent reviews of the events were conducted to verify the accuracy and completeness of licensee information.

a. Inoperable Standby Liquid Control System (SLCS)

On June 2, 1988, the licensee discovered during a routine quarterly surveillance test that both SLCS pump discharge accumulators were pressurized below the minimum specified pressure. There is one accumulator on the discharge piping of each of the positive displacement pumps which dampens pulsations. Each is a small steel vessel with a synthetic bladder. The bladder's upper side should be charged with nitrogen to about 500 psig. Both accumulators were found with less than 300 psig nitrogen pressure. Peak to peak pressure fluctuations of approximately 30-40% of discharge pressure can occur with zero accumulator pressure. For an accumulator pressure of 300 psig, the pressure fluctuations are reduced to 15-20% of the discharge pressure, and the fluctuations would be approximately 5% at 500 psig. Excessive pressure fluctuations can result in overstressing of the discharge piping. The licensee's analysis indicates that the SLCS could be considered operable with the accumulator pressure above 300 psig. Based on the as-found pressures, the licensee determined that the system was inoperable and notified the NRC via the ENS on June 3, 1988. A leak test found that the nitrogen addition valves on the accumulator were leaking past the valve stem. An inspection of the valves indicated that the failure was due to normal wear. The licensee replaced the bladders and the accumulators were fully recharged.

b. Inadvertent Reactor Scram During Instrument Calibration

At 12:52 p.m. on July 5, 1988, the licensee experienced an inadvertent reactor scram during the performance of instrument calibration procedure 3.M.2-5.2, "Intermediate Range Monitor (IRM) Calibration". The licensee's investigation indicated that the scram resulted from spurious spiking of "A" and "B" IRM channels when the high voltage power supply for the "E" IRM channel was inserted. The "E" IRM drawer is located between "A" and "B" IRM drawers. Installation of the power supply caused the spiking due to the close proximity of the three units. The licensee determined that the root cause was a procedural error allowing the insertion of the high voltage power supply without prior removal of the voltage preregulator. Procedure 3.M.2-5.2 was last revised on February 5, 1988 to rearrange the test sequence. That revision inadvertently deleted the step in the procedure which requires removal of the voltage preregulator. The procedure was subsequently revised.

c. Unexpected Reactor Scram During ATWS Testing

On July 8, 1988 at 2:52 p.m. an unexpected scram occurred during performance of a post-modification test TP87-126 on the Anticipated Transient Without Scram System (ATWS). The test procedure included the intentional generation of an ATWS alternate rod insertion (ARI) signal. The ARI signal caused the scram inlet and outlet valves to open, and the scram discharge instrument volume (SDIV) vent and drain valves to close. The control rod drive pumps had been removed from service, however the standing head of water in the reactor vessel combined with the effect of the ARI signal caused water to accumulate in the SDIV. The SDIV high water level scram setpoint was reached, generating an unexpected scram signal. Neither the Modification Management Group responsible for development and implementation of test procedure TP87-126, nor the Operations Department personnel releasing the activity, had identified the potential scram before its occurrence. The actuation was reported to the NRC via ENS at 3:25 p.m. on July 8, 1988. In response to this event, the licensee halted the test and reviewed the procedure to identify any additional scrams or engineered safeguards feature actuations. No additional actuations were identified.

d. Unexpected Emergency Diesel Generator Start During Implementation of Equipment Isolations

On July 12, 1988, at 9:30 p.m. an unexpected start of the "B" Emergency Diesel Generator (EDG) occurred during isolation of the associated 4160 VAC bus, A6. Operators implementing an approved tagging sheet had opened and racked down the A6 feeder breaker from the unit auxiliary transformer. The feeder breaker to A6 from the startup transformer was also opened and the bus deenergized. When the startup transformer breaker was racked down its breaker cell position switch closed as designed. Closure of the cell position switch in combination with a unit auxiliary transformer breaker open signal initiated an automatic start of the "B" EDG. The EDG output breaker had previously been removed from service so the generator did not close onto the bus. Neither the Maintenance Department personnel responsible for planning the activity nor the Operations Department personnel approving the work for implementation, had identified the likelihood of the EDG engine actuation before its occurrence. The actuation was reported to the NRC via ENS at 10:25 p.m.

An identical unanticipated start of the "B" EDG occurred in September, 1987 due to a deficient surveillance test procedure. This actuation was caused by the same sequence of equipment operations described above, as detailed in inspection report 50-293/87-45. In addition, in February 1988 Notice of Violation 50-293/88-07-01 was issued when the inspector identified that an approved test procedure, if implemented, would have caused unanticipated EDG starts, again by the same sequence of equipment operations. The inspector expressed concern to licensee management that inadvertent actuations caused by failure to recognize the effects of this same equipment operation had occurred three times in a ten-month period, indicating that implemented corrective actions had not been effective. Three ESF actuations occurred during the current inspection period. During the past seven months the licensee has experienced 19 ESF actuations. In several cases similar actuations had previously occurred. In response to Violation 88-07-01 the licensee formed a task force to evaluate the underlying causes for the frequency of engineered safety feature actuations at Pilgrim. This task force evaluation is continuing. The licensee effort to identify the root causes for, and to reduce the number of ESF actuations will be reviewed during a future inspection. This area will remain unresolved pending completion of licensee evaluations (UNR 88-25-02).

e. Licensee's Press Release Concerning a Study Done by the Massachusetts Department of Public Health (MDPH)

On July 1, 1988 the licensee informed the NRC via ENS in accordance with 10 CFR 50.72 (b)(2)(vi) that a press release related to matters potentially affecting the public health and safety was planned. The press release provided the licensee's interpretation of a study done by the MDPH on wind patterns around the Pilgrim Station and their possible effects on leukemia rates in surrounding towns.

5.0 Review of LER's

LER's submitted to NRC:RI were reviewed to verify that the details were clearly reported, including accuracy of the description of cause and adequacy of corrective action. The inspector determined whether further information was required from the licensee, whether generic implications were indicated, and whether the event warranted onsite followup. The following LER's were reviewed:

LER No.	Event Date	Subject
87-19-00	12/27/87	Unplanned actuation of the primary containment (Group 6) isolation system. The cause of the actuation was an instrument and control technician error during installation of a temperature switch. Immediate inspector followup of the actuation is described in inspection report 50-293/87-57.
87-20-00	12/17/87	Failure of fire dampers to close during a surveillance test. The cause was due to incorrect installation of the dampers. The inspector's review of the LER is described in inspection report 50-293/87-57.
87-21-00	04/28/87	Automatic start of "A" emergency diesel generator due to an inadvertent emergency core cooling system initiation signal. The cause was due to improper venting of the drywell pressure transmitters during a post-modification test. The inspector's followup of this actuation is described in inspection report 50-293/87-18.
87-22-00	06/07/87	Secondary containment isolation signal due to failed relay. The cause of the actuation was a random failure of a relay in the "B" refueling floor vent exhaust radiation monitor trip circuit. The failed relay was manufactured by Potter and Brumfield, model/type KHS 17013, 24 volt DC coil. The inspector followup of the actuation is documented in inspection report 50-293/87-26.

LER No.	Event Date	Subject
87-23-00	09/17/87	Automatic actuation of the "A" emergency diesel generator during a surveillance test. The cause of the actuation was a procedural deficiency. The diesel start circuit includes an initiation signal when the bus feeder breaker from the startup transformer is racked down, while the bus feeder breaker from the unit auxiliary transformer is open. Procedure 3.M.3-1, "A5/A6 Buses 4KV Protective Relay Calibration/Functional Test", did not identify this diesel start. Immediate followup of the actuation is documented in inspection report 87-45. A similar auto-actuation of a diesel generator occurred on July 12, 1988 during isolation of the associated 4KV bus. The details of this actuation are described in section 4.d of this report.
87-24-00	10/06/87	Loss of power to a 480 volt AC emergency bus due to a breaker trip. No definite cause for the spurious breaker trip was identified. The licensee postulated that individuals working in the area may have inadvertently caused the trip. The inspector followup of this event is described in inspection report 50-293/87-45.

In inspection report 50-293/87-45 the NRC identified three engineered safety feature actuations which appeared to be reportable under 10 CFR 50.73 but had not been reported by the licensee. The licensee reviewed the three actuations, agreed that they should have been reported and issued Licensee Event Reports (LER) 87-010, 87-011, and 87-012 to document the occurrences. In addition the licensee reviewed previous actuations to determine if any additional reports were needed. Subsequently, the licensee submitted LERs 87-021. 87-022, 87-023 and 87-024, discussed above, as a result of their review.

6.0 Management Meetings

At periodic intervals during the course of the inspection period, meetings were held with senior facility management to discuss the inspection scope and preliminary findings of the resident inspectors. A final exit interview was conducted on July 26, 1988. No written material was given to the licensee that was not previously available to the public. At no time during the inspection did the licensee identify any materials provided for review which contained proprietary information.

On June 27, 1988, a publicly held management meeting was conducted at the NRC Region I Office in King of Prussia, Pennsylvania, to discuss the licensee's response to NRC Maintenance Team Inspection 50-293/88-17, and their actions in followup of their self-assessment of restart readiness. At that meeting, the licensee requested commencement of a comprehensive NRC Integrated Assessment Team Inspection (IATI). The meeting was attended by Assistant Secretary of Public Safety, Commonwealth of Massachusetts, Mr. Jeffrey Hausner, Director, Nuclear Safety Emergency Preparedness Program, and Mr. George Dean, Assistant Attorney General, of Commonwealth of Massachusetts were also in attendance. The meeting is described in NRC Region I Meeting Report No. 50-293/88-26.

Attachment I to Inspection Report 50-293/88-25

Persons Contacted

R. Bird, Senior Vice President - Nuclear
K. Highfill, Site Director
* R. Anderson, Plant Manager
E. Kraft, Plant Support Department Manager
A. Morisi, Acting Outage and Planning Manager
D. Swanson, Nuclear Engineering Department Manager
J. Alexander, Operations Section Manager
J. Jens, Radiological Section Manager
J. Seery, Technical Section Manager
R. Sherry, Maintenance Section Manager
P. Mastrangelo, Chief Operating Engineer
D. Long, Security Division Manager
W. Clancy, Systems Division Manager
F. Wozniak, Fire Protection Division Manager

*Senior licensee representative present at the exit meeting.