U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.: 50-293

Report No.: 50-293/92-04

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

Facility: Pilgrim Nuclear Power Station

Location: Plymouth, Massachusetts

Dates: March 17 - May 4, 1992

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Approved by: FOR 5/27/92
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Inspection Summary:

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

Results: Inspection results are summarized in the Executive Summary.

Violation: A non-cited violation (NCV) was identified regarding inadequate review and implementation of a temporary procedure which resulted in an unplanned partial actuation of the emergency core cooling system logic.

Unresolved Item: Two unresolved items were identified:

The first unresolved item concerns licensee reanalysis of seismic design margins for safety related piping systems affected by misapplication of the NRC guidance on the use of damping ratios (Section 7.2.1, UNR 50-293/92-04-01).

The second unresolved item concerns continuing licensee evaluation of reactor vessel water level instrumentation spiking and potential contribution to the spiking due to the presence of non-condensible gasses (Section 8.2.2, UNR 50-293/92-04-02).

EXECUTIVE SUMMARY

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Plant Operations: Excellent command and control skills were exhibited by control room operators during the March 26 reactor shutdown. Operations Section Management provided clear guidance to the operators regarding reactor core isolation cooling system valve repair and potential emergency action level entry conditions.

Outage activities were well coordinated. However, operator error, with the reactor shutdown and all control rods fully inserted, caused an unplanned reactor protection system actuation.

Operators demonstrated excellent communications and interdepartmental coordination during reactor vessel level instrumentation post modification testing. Of particular note was the outstanding oversight of reactivity manipulations. Restoration of the augmented offgas system following reactor startup was well controlled.

Radiological Controls: Radiation Protection (RP) personnel response to a worker contamination event was timely and appropriate. Additional testing to verify that no internal overexposure had occurred was conducted and licensee actions to evaluate this event for RP lessons learned and improved work controls were properly oriented.

Maintenance and Surveillance: Inspection and repair of motor operated valve MO-1301-16 were timely. The associated failure analysis team effectively identified the cause of the valve operator malfunction and initiated well focused corrective actions.

Emergency Preparedness: Followup on the communication of backup meteorological data from the licensee emergency operational facility to the Commonwealth of Massachusetts emergency operations center during the December 1991 emergency exercise supported the identification of this area as an area for improvement. A practice emergency preparedness drill was conducted on April 9, 1992. The licensee drill critique was an effective self assessment tool. In addition, use of the simulator provided a more realistic environment in which to initiate and evaluate the effectiveness of operator response to an event.

Security: Security personnel continued to perform assigned duties in an effective manner.

Safety Assessment and Quality Verification: The licensee performed several inspections of motor operated valve (MOV) torque switches in response to a 10 CFR 21 notification. Four problem torque switches were identified and promptly replaced. Direction of inspection effort was well focused toward those safety-related MOVs with the highest likelihood of containing problem torque switches.

(EXECUTIVE SUMMARY CONTINUED)

Engineering and Technical Support: Thrust testing of several safety-related MOVs was conducted in response to safety concerns identified by a recent NRC team inspection. Testing was conducted in a highly professional manner and was timely with respect to the identified concerns. Investigation and causal analysis of the continuing reactor vessel water level instrumentation spiking, experienced during reactor depressurization, was generally well controlled and reflected sound safety perspectives. Appropriate resources and technical expertise were provided to the root cause analysis team. The request for waiver of compliance and associated safety evaluation were effectively supported by design bases documentation. Subsequent post modification testing was conducted in a deliberate manner. Notwithstanding, inadequate reviews of a deficient temporary procedure caused an unplanned actuation of portions of the emergency core cooling system logic.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

At the beginning of the report period, Pilgrim Nuclear Power Station (PNPS) was operating at approximately 100% of rated power.

On March 18, 1992 the control room full core display experienced full in and full out light indications for several control rods on the right side of the display. Licensee evaluation concluded the most likely cause for the observed light indications was the presence of moisture under the reactor vessel.

On March 23, the reactor core isolation cooling system (RCIC) was declared inoperable in order to replace the turbine control EG-R, to overhaul the RCIC full flow test valve (RCIC-53) motor actuator, and to perform miscellaneous maintenance activities. On March 25 during post maintenance testing, inside containment isolation valve RCIC-16 (steam supply line) failed to stroke full closed.

In accordance with Technical Specifications, outside containment isolation valve RCIC-17 was maintained closed which precluded further RCIC testing. Later on March 25, a controlled reactor shutdown was initiated for the purpose of entering into an eight day maintenance outage during which period the RCIC-16 valve malfunction would be investigated.

On March 26-27, during reactor vessel depressurization, three Group I primary containment isolation system (PCIS) actuations occurred due to reactor vessel water level instrumentation spiking. A licensee root cause analysis team (RCAT) was chartered to investigate these actuations (Section 8.2).

In addition to repair of the RCIC-16 valve and evaluation of the reactor vessel water level instrumentation spiking events, significant outage activities included repair of a body to bonnet leak on the RCIC-17 valve, replacement of seals on two control rod drives, testing of NRC Generic Letter 89-10 applicable motor operated valves, and repair of a main turbine control valve and the main turbine thrust bearing.

On April 9, reactor startup was initiated for the purpose of conducting "B" reference leg post modification testing. The startup and testing were conducted in accordance with the conditions of a Regional Waiver of Compliance that had been granted on April 8. The testing, which included reactor startup to approximately 12% of rated power, reactor vessel metal temperature soak to equilibrium, and subsequent reactor shutdown, depressurization, and entry into shutdown cooling was completed satisfactorily on April 11.

On April 12, reactor startup was commenced. The main generator was synchronized to the offsite distribution system at 9:07 am on April 13 and 100% of rated power was achieved on April 16.

At the conclusion of the inspection period, the station was operating at 100% of rated power.

2.0 PLANT OPERATIONS (71707, 40500, 90712)

2.1 Plant Operations Review

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

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

Control ro instruments were observed for correlation between channels, proper functioning and conformance with Technical Specifications. Alarms received in the control room were reviewed and discussed with the operators. Operator awareness and response to these conditions were reviewed. Operators were found cognizant of board and plant conditions. Control room and shift manning were compared with Technical Specification requirements. Posting and control of radiation, contamination, and high radiation areas were inspected. Use of and compliance with radiation work permits and use of required personnel monitoring devices were checked. Plant housekeeping controls, including control of flammable and other hazardous materials, were observed. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout, and lifted lead and jumper logs. Inspections were performed on backshifts including March 17-20, 23-27, 30 and April 1-3, 6-8, 13-17, 21-24, 27. Deep backshift inspection was performed during the following periods:

April 10 (10:00 pm - 12:00 am)	April 11 (12:00 am - 3:15 am)
April 12 (5:30 pm - 10:30 pm)	April 20 (7:10 am - 2:35 pm)

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

2.2 Controlled Reactor Shutdown

On March 25, the licensee initiated reactor core isolation cooling system (RCIC) post maintenance testing following a brief two day system outage to replace the governor EG-R, to overhaul the full flow test valve, and for miscellaneous maintenance activities. During testing, the RCIC steam supply inside containment isolation valve (RCIC-16) failed to stroke properly. The associated outside containment isolation valve, RCIC-17, was closed and deenergized in accordance with Technical Specification (TS) requirements. Closure of RCIC-17 precluded further system testing. Initial troubleshooting of RCIC-16 indicated motor operator malfunction.

Because the valve is located inside the drywell, a reactor shutdown was necessary in order to accomplish repairs. In conjunction with the repair of RCIC-16, the licensee elected to enter into a planned eight day maintenance outage.

The inspector reviewed licensee shutdown plans and projected RCIC-16 repair scope with respect to TS requirements and emergency action level entry conditions. The licensee effectively analyzed the most probable RCIC-16 failure mechanism and established with reasonable engineering judgement a bounding repair time estimate. Additionally, operations section management issued a memorandum, via night order, to control room operators that provided concise guidance and instruction for potential situations that could present EAL entry conditions. The inspector determined the RCIC-16 repair projection and accompanying night order memorandum were appropriately supported by TS.

2.3 Group I Isolations During Reactor Depressurization

On March 26 and 27 with the reactor mode select switch in shutdown, three automatic Group I primary containment isolation system (PCIS) actuations occurred due to sensed or [illegible] reactor vessel water levels. The PCIS Group I high reactor vessel water level TS setp[illegible] inches with an actual calibration setpoint of +45 inches. The first and third isolations [illegible] to spiking on reactor vessel water level instrumentation and are discussed in Section 8.2.

The second isolation occurred on March 26, at 9:29 pm during recovery from the initial Group I isolation and was the result of an actual increasing reactor vessel water level.

Following the first isolation, the reactor vessel was stabilized at +29 inches water level and approximately 82 psig pressure. In accordance with procedure 2.2.92, "Main Steam Line Isolation and Turbine Bypass Valves," control room operators reset the initial isolation, opened the outside containment main steam isolation valves (MSIVs), equalized reactor pressure and steam header pressure to within 50 psig, and opened the "D" main steam line inside containment MSIV. Reactor vessel water level experienced a momentary swell from the initial +29 inches to approximately +46 inches upon opening of the "D" main steam line MSIV. Reactor vessel water level instrumentation responded to the level swell and the "A" side reference leg instrumentation initiated the Group I isolation. All PCIS components responded to the actuation as designed. The isolation was reset and the MSIVs and main steamline drain valves were reopened on March 27 at 12:25 am.

Licensee causal analysis of this event concluded that the initial reactor vessel water level of +29 inches, while well within normal parameters, was too high to accept the level swell incumbent with reestablishment of direct reactor vessel and main steam header communication. The licensee revised procedure 2.2.92 to instruct control room operators to reduce reactor vessel water level to the lower end of the normal level range, not to exceed +24 inches prior to reestablishing direct reactor vessel and main steam header communication.

Inspector review concluded licensee response to this event was appropriate. The cause of the isolation was identified and understood before being reset. Subsequent procedure revisions were promptly completed. The inspector had no further questions regarding this event.

2.4 Augmented Offgas (AOG) System Pressure Transient

On April 14 with the reactor at power and 1000 psig pressure, operator inspection of the condenser bay area detected a small steam leak from an AOG pressure instrumentation sensing line isolation valve (1-HO-16). The AOG system was isolated and the valve was repacked. Isolation for the repair included closure of pressure control valves (PCV) 9238 and 9239 which provide a regulated steam supply to the AOG steam jet compressor. Several hours later on April 14, following repair of 1-HO-16, operations personnel commenced restoration and startup of the AOG system in accordance with procedure 2.2.106, "Augmented Offgas System".

As PCV-9239 was placed in service the valve appeared to operate erratically, quickly positioning to the full open position. A resultant momentary pressure pulse was sensed at the inlet to the offgas holdup line which caused the four main condenser vapor valves to shut upon sensing pressure in excess of 35 psig. Closure of the vapor valves isolated the path by which noncondensable gases are extracted from the main condenser, treated by the AOG system and subsequently released through the main stack. Control room operators quickly determined the cause of the vapor valve isolation, verified AOG system parameters, and reopened the vapor valves. Their prompt response demonstrated a high level of alertness to plant conditions.

The activity level of the main stack gaseous effluent peaked immediately after the vapor valves were reopened. The peak effluent value of approximately 2400 counts per second lasted for less than a minute. Utilizing the guidance of NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," the licensee determined the peak release rate to be less than 20 percent of that permitted by Technical Specifications. The thirty minute delay line and offgas filters were returned to service upon reopening of the vapor valves. Main stack gaseous effluent activity remained less than one percent of Technical Specification limits for the remainder of the day. The AOG portion of the offgas system remained isolated for engineering evaluation until late in the day. The licensee maintained reactor power level below 50 percent, as required by Technical Specifications, until after the AOG was restored to operation. Reactor building and main stack effluent activity levels were monitored closely and remained well below Technical Specification limits throughout the day. The total combined reactor building and main stack effluent activity released on April 14, 1992, resulted in a site boundary dose equivalent of approximately two percent of that permitted by Technical Specifications. The inspector reviewed licensee calculations of the peak instantaneous, maximum hourly, and daily effluent activity release for April 14, 1992. Calculations were conservative and verified that effluent activity remained well below Technical Specification limits.

Offgas system piping included a rupture disk initially intended to rupture at 40 psig to minimize system damage from pressure transients. The licensee conducted a valve lineup and a system walkdown. The rupture disk was found breached which resulted in a small amount of noncondensable gasses being released to the condenser bay. Condenser bay ventilation is discharged via the monitored reactor building exhaust stack. Vendor review of industry operating experience determined that the rupture disk was not needed to protect system integrity and could be removed from the offgas system. Operations personnel isolated the rupture disk and completed the offgas system walkdown. No further abnormal conditions were observed. Both AOG hydrogen analyzers were calibrated to ensure hydrogen sensors were not damaged by the pressure excursion. Functional evaluation of PCV-9238 and 9239 indicated that valve response was sluggish during pressure regulator startup, but acceptable for operation. The walkdown of the offgas system and regulator evaluations were thorough.

Problem Report 92-9025 was initiated to determine the cause and effects of the pressure pulse in the AOG system. Preliminary licensee evaluation identified the primary cause to be an inadequate procedure. Procedure 2.2.106 did not address placing PCV-9238 and 9239 in service with the steam supply line already pressurized. Routine system startup typically involved aligning the PCVs before steam supply pressure was available. In this case however, the PCVs were placed in service following repair of 1-HO-16 with the steam supply pressurized. The sluggish response of PCV-9238 and 9239 was determined to be a contributing cause. The nuclear watch engineer authorized a temporary procedure change (SRO change 92-31) to address placing the pressure regulators in service with steam supply pressure available. An engineering evaluation determined that AOG operation with the rupture disk isolated would not adversely impact AOG system components or performance. The AOG system was returned to operation early in the evening of April 14, 1991. The licensee identified several long-term corrective actions including training and permanent revision to procedure 2.2.106, and evaluation of PCV-9238 and 9239 for replacement. The detailed investigation and assessment conducted prior to AOG restoration demonstrated a sound licensee safety perspective.

3.0 RADIOLOGICAL CONTROLS (71707)

3.1 Personnel Exposure/Intake Evaluation

On April 10, 1992, with the primary containment de-inerted and available for personnel access as a result of the need to conduct RCIC steam supply line inside containment isolation valve repairs, additional maintenance activities were in progress in the drywell. One of the items of work in preparation for further plant power ascension was the injection of a leak sealant (i.e., Furmanite) into the "B" recirculation pump (P-201B) jacking bolt holes to seal a suspected area of minor RCS leakage. Two contractor workers, accompanied by a licensee radiological protection (RP) technician, entered the drywell in accordance with a radiation work permit (RWP No. 92-336) which required protective clothing, including face shields, for the work crew. Respirators were not specified because of the hot environment and the low probability of airborne activity based upon the history of previous Furmanite injections into this pump.

After completion of the work and upon exit from the drywell, both contractors alarmed the radiation portal monitor at elevation 23' inside the reactor building. The air sample lapel was checked, a backup air sample was taken, and the workers and RP technician were dispatched to the radiologically controlled area RP checkpoint for decontamination. The backup air sample results indicated no unacceptable levels or residual airborne activity in the area of recirculation pump P-201B. All three personnel were escorted by RP for whole body count testing, with initial results indicating a possible radiological uptake by the primary Furmanite worker in excess of the quarterly limits allowed by 10 CFR 20.103.

The cause of the contamination and exposure appears to be the release of a small volume of reactor coolant, which flashed to steam when a Furmanite fitting was removed from the pump. Subsequent radionuclide intake evaluation for the one worker suspected of an uptake in excess of quarterly maximum permissible concentration (MPC) limits by Yankee Atomic Electric Company determined that no overexposure had occurred. The YAEC test results for the lungs, GI tract, and the thyroid for various radionuclides indicated an internal exposure of the individual well below the reportable limits of 10 CFR 20.

The inspector was informed of this personnel contamination event by a cognizant RP section manager within a few hours of the occurrence, and before the final YAEC bioassay results were available. Licensee understanding of the reporting requirements of 10 CFR 20.405 was confirmed. Subsequently, the licensee notified the inspector of the final YAEC intake evaluation results and the determination that no formal report to the NRC was required. The inspector reviewed problem report 92-0222 documenting this personnel contamination event, its radiological evaluation, to date, and the subject air sample and worker uptake test results. It was noted that the licensee had properly roped off the contamination area near pump P-201B until backup air sample data verified that airborne radioactive material concentrations were within acceptable levels. The inspector discussed this event and the associated problem report with the Radiological Section Manager and determined that the licensee had conducted its followup and evaluation of this issue in accordance with station procedures and prudent assessment techniques. The licensee is further evaluating the lessons learned from this event from an RP perspective. The licensee agreed with the inspector that indepartmental coordination, regarding any future Furmanite injection activities at similar plant reactor coolant system pressure and temperature conditions, would be prudent from a maintenance, as well as RP standpoint.

The inspector has no additional questions and considers licensee followup actions, to date, to be both acceptable and oriented toward preventing a repetitive similar event.

3.2 Interim Storage of Low Level Radioactive Waste Update

Licensee plans to establish an onsite storage facility for low level radioactive waste were previously reviewed in NRC Inspection Report 50-293/92-02. The licensee completed evaluation of bids for the design of the facility and has issued the design contract.

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

4.1 Repair of the Reactor Core Isolation Cooling (RCIC) Steam Supply Valve

On March 25, 1991, the RCIC steam supply valve (MO-1301-16) failed to close during system restoration following unrelated system maintenance. Position indication in the control room showed the valve to be closed, but steam supply line pressure indicated that the valve was not fully seated. Following de-inerting of the drywell, the inspector accompanied maintenance personnel to perform a visual inspection of MO-1301-16. The motor operator was to have found separated from the yoke and had travelled up the valve stem. The four capscrews which attach the yoke to the operator had backed out of the sockets allowing the stem nut to drive the motor operator up the stem. Brass filings from a damaged stem nut were present. However, no further visible damage was apparent. The maintenance engineer promptly informed operations personnel of apparent valve condition and estimated repair time. This information supported determination that valve repair and system restoration could be completed within the period specified by Technical Specifications. Repairs were completed during a subsequent plant shutdown during which the licensee performed several planned maintenance activities. Initial inplace inspection and assessment of MO-1301-16 damage was timely and provided excellent information important to plant operation.

The motor operator for MO-1301-16 was removed, inspected, and overhauled. The stem nut was verified to be the only damage to the operator and was replaced. The motor operator was reinstalled in accordance with procedure 3.M.3-24.1, "Limitorque Valve Operator Removal and Reinstallation." The inspection of the operator was thorough and provided valuable information for subsequent root cause analysis. As a design enhancement, the original 1 inch long capscrews were replaced with 1.5 inch long capscrews (The 1 inch capscrews had provided only 1/8 inch engagement into the operator) and were torqued to 30 foot pounds as specified in the vendor manual. Diagnostic and local leak rate testing were performed to verify valve operability and integrity of the valve internals. Testing was properly conducted with satisfactory results prior to returning the RCIC system to service.

The licensee established a failure analysis team to evaluate the cause of the MO-1301-16 malfunction. The team determined that the motor operator separated from the valve yoke because the four capscrews had been insufficiently torqued. The capscrews had been torqued to 8 foot pounds during the previous refueling outage vice 30 foot pounds as specified in the vendor manual. The root cause of the incorrect torque value was attributed to errors in procedure 3.M.3-24.1 and drawing M137A-1. The licensee implemented corrective revisions to the procedure and the drawing to address the noted errors. In addition, a separate problem report was initiated to address the generic issue of potential inadequate torquing of yoke to operator fasteners. A verification capscrew torque check was performed on several additional safety-related motor operated valves (MOVs) prior to reactor startup. The torque checks were completed satisfactorily with no operability concerns identified. The root cause evaluation was detailed. Corrective actions were thorough and appropriate to address both operability of MO-1301-16 and the generic impact on other MOVs within the plant.

4.2 Recirculation Pump M-G Set Brush Replacement

As documented in NRC inspection report 50-293/92-01, the "A" recirculation pump motor generator (M-G) set was secured on January 29, 1992 to facilitate the replacement of the electrical brushes at the generator end of the M-G set. Approximately 24 hours later, the "A" M-G set was again secured to install a temporary modification on the rigging for the brush holder device. Both of these maintenance activities involved reactor power reductions to levels in the range of 40-45 percent and single loop operation under the constraints imposed by the PNPS license conditions.

On February 14, 1992, the licensee issued revision 7 of procedure no. 3.M.3-7 which added a new section 8.6, detailing the steps for the replacement of the recirculation pump M-G set brushes with the M-G set equipment remaining in operation. This procedure change was reviewed and recommended for implementation by the station operations review committee (ORC) during a special ORC meeting, no. 92-12, convened on February 14, 1992. On February 24, 1992, the "B" recirculation pump M-G set brushes on the exciter end were replaced on-line in accordance with the new procedural provisions. The procedural steps of section 8.6 include a cautionary statement regarding personnel safety and protective equipment while working on live electrical equipment.

During this inspection period, with the reactor shutdown for the forced outage, the licensee replaced the electrical brushes and brush holders on the exciter and generator ends of both the "A" and "B" recirculation pump M-G sets. A new brush holder design was selected for its ease of installation, facilitating safer brush replacement activities with the M-G set remaining on-line. The new brush holder design was verified by the licensee to have been installed at other BWR facilities where on-line replacement was either used or planned.

The inspector reviewed section 8.6 of procedure no. 3.M.3-7, revision 7, and the special ORC meeting no. 92-12 report. The inspector also verified that the M-G set brush holder replacement activities conducted during this inspection period were governed by approved maintenance requests, i.e., MR 19103958 for the "A" M-G set and MR 19105108 for the "B" M-G set. The inspector determined that the new procedural controls for the replacement of the recirculation pump M-G set brushes on-line with the equipment in operation were adequate for the task. The installation of the new brush holders during this inspection period generally facilitates the brush replacement activities. Licensee actions to implement such controls and provide for a brush holder design with improved brush change-out flexibility have appropriately addressed a maintenance requirement while considering the impact on operations. The inspector identified no unresolved safety concerns and has no further questions on the conduct of these maintenance activities.

5.0 EMERGENCY PREPAREDNESS (40500)

5.1 Practice Emergency Preparedness Drill

A practice emergency preparedness (EP) drill was conducted on April 9, 1992. The following emergency response facilities were activated to participate in drill response: Technical Support Center, Operations Support Center, Emergency Offsite Facility (EOF), and the Plymouth and corporate media centers. The EP drill utilized the control room simulator for drill initiation and response. The inspector observed portions of the drill at both the control room simulator and the EOF. Use of the simulator provided an increased level of realism to control room personnel by which their actions could be evaluated. Emergency plan implementing procedures were effectively utilized for emergency action classification. The nuclear operations supervisor demonstrated excellent command and control throughout implementation of emergency operating procedure actions. Status briefings conducted at the EOF were clear and contained an appropriate level of detail.

The inspector reviewed the licensee critique of drill performance. Drill evaluator comments regarding areas of superior performance as well as areas for improvement were detailed and constructive. The breadth of comment scope demonstrated the licensee capability to perform effective self assessment.

5.2 Emergency Operations Facility (EOF) Information

During the December 1991 emergency exercise, the licensee form documenting an exercise General Emergency condition, as sent from the licensee Emergency Operations Facility (EOF) to the Commonwealth of Massachusetts Emergency Operations Center (EOC), noted that the meteorological tower was out of service. The backup meteorological tower data (scenario wind speed and direction) were available in the EOF, but were not included on the notification form transmitted to the EOC.

NRC followup found no specific transmission of scenario weather information from the licensee to the State. It was indicated that the State was provided the scenario weather by an exercise controller incident to initiation of a Commonwealth effort to obtain backup weather data. However, since the lead federal agency for review of off-site emergency preparedness performance is the Federal Emergency Management Agency (FEMA), pending issuance of the FEMA exercise report, the NRC's assessment of this aspect was limited to on-site licensee performance. Related information received by the NRC was provided to FEMA for consideration as appropriate.

In this case, and as is typical, the exercise plant operators were not the actual on-shift operators and the scenario weather was different from the actual weather. These factors necessitated an artificiality in the provision of backup weather information. NRC discussions with licensee emergency preparedness personnel confirmed the availability of backup (scenario) weather information to the EOF and its inclusion on data sheets for entry onto EOF status boards.

However, no record or specific recollection of posting of that information on the status boards or of communication of that information to Commonwealth responders was provided. The licensee expressed the following intentions.

- To include backup meteorological data on the notification forms when primary meteorological tower data is not available.

- To further discuss EOF information sources with responders, and to provide a written description of those sources for the responder information folders.

The communication of weather data during emergency conditions was classified as an emergency exercise area for improvement (i.e., a matter for licensee review to determine whether corrective action is appropriate). Licensee actions will be reviewed during routine inspection, and effectiveness of the licensee's measures will be assessed during the next emergency exercise.

6.0 **SECURITY** (71707)

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

7.0 **SAFETY ASSESSMENT AND QUALITY VERIFICATION** (92701)

7.1 **Limitorque Torque Switch Fiber Spacer, 10 CFR 21 Review**

In September of 1989, Limitorque Corporation issued a 10 CFR 21 notification regarding SMB-00 and SMB-000 cam-type torque switches with fiber spacers under the electrical contact bridge. The fiber spacer represented a potential common mode failure which could adversely effect the balance of the torque switch and resultant performance characteristics. Torque switch design had been modified in 1976 (SMB-00) and 1980 (SMB-000) to correct the problem by replacing the fiber spacers with a metallic contact bridge. The 10 CFR 21 notification recommended replacement of older SMB-00 and SMB-000 containing fiber spacers with the newer design during the next available maintenance period. The licensee initiated a review (PCAQ 89-122) to evaluate the impact of this concern with regard to torque switches installed or maintained as spare parts at Pilgrim Station. This review narrowed the scope of applicability, but did not complete an evaluation of all installed SMB-00 and SMB-000 torque switches throughout the plant.

The licensee utilized the March 26, 1992 maintenance outage as an opportunity to perform Part 21 inspections of several safety-related MOV torque switches. A maintenance history review and safety function assessment was performed to prioritize those MOVs which remained to be

Inspection of nine MOVs identified four torque switches which contained fiber spacers. Those four were replaced with torque switches of the recommended design. The remaining eleven torque switch inspections have been scheduled for completion during the upcoming midcycle outage. Identification and inspection of the nine MOVs during this maintenance outage demonstrated a sound safety perspective.

7.2 Licensee Event Report (LER) Review

7.2.1 LER 92-01

LER 92-01, "Class I Piping Seismic Damping Ratios," dated February 21, 1992, is a voluntary report describing a potential decrease in the design margins of safety for piping and piping support response to seismic events. Due to a misinterpretation of the NRC guidance on seismic damping rations delineated in Regulatory Guide 1.61 and the misapplication of ASME Code Case N-411 without adoption of the seismic spectra consistent with NRC endorsement of this Code Case, higher damping values than allowed were applied to plant design changes. Despite these errors, no adverse impact on the affected Class I piping and supports was identified. Licensee analysis of these conditions consistent with NRC Generic Letter 91-18, indicated that all affected plant systems remain operable. The inspector reviewed the operability evaluation, attended the ORC meeting (92-03) recommending its approval, and verified that the operability determination is appropriately based upon the allowable stress limits governed by the ASME Code and NRC generic guidance (e.g., Bulletin 79-14).

Licensee corrective actions include the identification and reverification of existing stress analyses, affected by the decreased margin to material yield strength, and the systematic review of all safety-related piping systems for which the seismic design margins currently are in question with respect to regulatory guidelines. These corrective actions are currently being tracked by the following licensee documents: Potential Condition Adverse to Quality (PCAQ 92-3) and Management Corrective Action Request (MCAR 92-1). While the inspector has no additional questions regarding the immediate licensee corrective measures relative to the identified nonconforming conditions or the related operability evaluation, it is noted that interim plant operations, in accordance with generic NRC guidance, is limited to the current operating cycle. Thus, pending completion of the planned licensee reanalyses and demonstration that existing piping/pipe supports conform to the appropriate design specifications and requirements, the nonconforming conditions identified in this LER remain unresolved (UNR 50-293/92-04-01).

While this LER was voluntary, it nevertheless considered the reporting requirements and addressed all appropriate criteria. Therefore, this LER is closed. However, due to operability constraints, recognized by the operability evaluation approved during special ORC meeting No. 92-03, successful licensee resolution of the technical and design margin issues relative to this LER and the above open item is required to be complete prior to plant restart subsequent to RFO No. 9.

7.2.2 LER 92-02

LER 92-02, "Inadvertent Actuation of a Portion of the Secondary Containment System during Surveillance Testing due to Limited Access to an Actuating Relay," dated March 19, 1992, describes the February 27, 1992, inadvertent actuation of the Channel "A" portion of the reactor building isolation control system (RBIS). Limited physical access contributed to the unintentional energization of the RBIS channel "A" RPWA relay by instrumentation and control technicians which resulted in the automatic closure of secondary containment system/reactor building train "A" supply and exhaust dampers and the automatic start of train "A" of the standby gas treatment system. This event was documented in NRC Inspection Report 50-293/92-03. This LER correctly addressed the reporting criteria. This LER is closed.

7.2.3 LER 92-03

LER 92-03, "Reactor Core Isolation Cooling (RCIC) System Made Inoperable per Technical Specifications due to an Inoperable Primary Containment System Isolation Valve," dated April 24, 1992, describes the March 25, 1992, RCIC steam supply valve failure to close. The RCIC system was correctly declared inoperable and the reactor was shutdown to support repair of the valve. This event is further documented in Sections 2.3 and 4.1 of this report. The LER accurately detailed the event, cause contributions and corrective actions. This LER is closed.

7.2.4 LER 92-04

LER 92-04, "Three Automatic Group I Isolations During Plant Shutdown," dated April 27, 1992, describes the PCIS actuations that occurred on March 26 and 27 during reactor depressurization. The isolations are documented in Sections 2.3 and 8.2 of this report. The LER effectively developed the individual isolation event descriptions in a logical manner and appropriately addressed the reporting criteria. This LER is closed.

7.2.5 LER 92-05

LER 92-05, "Unplanned Actuation of a Portion of Core Cooling Systems Logic Circuitry During Testing While Shutdown," dated May 4, 1992, describes the April 3, partial ECCS actuation due to implementation of an inadequate temporary procedure. This event is documented in Section 8.2.1 of this report. The LER provided excellent causal analysis of the event and appropriately addressed the reporting criteria. This LER is closed.

7.2.6 LER 92-06

LER 92-06, "Unplanned Scram Signal While Shutdown due to Licensed Operator Error," dated May 7, 1992, describes the April 11, unplanned reactor protection system (RPS) actuation signal that occurred with the reactor shutdown and all control rods fully inserted. The RPS scram signal occurred when the reactor mode select switch (RMSS) was moved from the shutdown position to the startup position with the scram discharge instrument volume (SDIV) level switches in a tripped condition.

Previous to the event, a planned reactor scram signal had been inserted to support undervessel maintenance. After completion of this work, the scram had been reset and SDIV had been drained, however the SDIV instruments, which are resistance temperature detectors (RTDs), had not reheated to normal temperatures and the SDIV high level scram signal remained present. Upon repositioning the RMSS from the shutdown position, where the SDIV high level scram is bypassed, to the startup position, the SDIV high level scram was unblocked and the RPS actuation occurred.

The cause of the event was the failure of licensed operators to ensure that the SDIV high level scram signal was cleared before repositioning the RMSS to a position where the scram signal is unblocked. The event was of no safety significance as the reactor was shutdown with all rods previously inserted. The LER effectively identified similar events and properly addressed the reporting criteria.

8.0 ENGINEERING AND TECHNICAL SUPPORT (71707)

8.1 Thrust Testing of Motor Operated Valve (MOV) 1301-53

The adequacy of torque switch settings and developed thrust for several safety-related MOVs was questioned during a recent NRC team inspection (50-293/92-80). The licensee performed diagnostic testing and reevaluated design thrust calculations for several valves, including RCIC full flow test valve MO-1301-53, in response to this concern.

The inspector observed valve operation test and evaluation system (VOTES) testing of MO-1301-53. Initial test setup required the valve to be manually placed off of the shut seat to remove pressure from the torque switch while a VOTES sensor was attached to the valve yoke. The licensee appropriately declared RCIC inoperable and notified the NRC. Although the system was inoperable, RCIC remained available for service if called upon to operate. Testing was conducted in accordance with procedure 8.Q.3-8, revision 12, "Limitorque Type SB/SMB Valve Operator Maintenance" and VOTES Users Manual (V-1152). During the initial open/closed cycle of the valve, recorded thrust shifted from compression to tension during the test. This "reversal" invalidated the test data and was attributed to VOTES sensor placement on the yoke. Orientation of the valve made proper sensor placement difficult to achieve. Following consultation with the vendor the licensee successfully relocated the sensor and obtained an "as-found" thrust value of 13,356 pounds. Engineering evaluation subsequently determined the minimum required design thrust to be 13,296 pounds. The valve was promptly returned to the fully closed position and the RCIC system was declared operable. Coordination of the evolution effectively maximized availability of the RCIC system. Communications and implementation of the test evolution were well controlled by the test supervisor. Test personnel demonstrated an excellent understanding of VOTES testing processes.

Visual inspection of the MO-1301-53 actuator during the testing evolution indicated a degradation of the grease lubricant and a twisted declutch shaft. The valve was locked in the fully closed position and the actuator was rebuilt. Maintenance personnel utilized this opportunity to install an improved type of grease lubricant and to upgrade the torque switch. Post work VOTES testing was performed as required by procedure 8.Q.3-8 following replacement of the torque switch. A loud ratcheting noise was heard upon cycling of the valve. The test coordinator immediately directed a prepositioned operator to open the power supply breaker to MO-1301-53. Prudent prepositioning of an operator and quick action by the test coordinator prevented damage to the actuator. Subsequent inspection determined that a metallic spacer designed to provide alignment between the worm shaft and the worm shaft clutch gear had not been reinstalled during actuator reassembly. The missing spacer resulted from miscommunication between technicians during reassembly. The maintenance supervisor properly addressed communications practices with maintenance technicians to preclude recurrence. Maintenance personnel inspected the actuator, replaced the missing spacer, and reassembled the actuator. The torque switch was adjusted during post work VOTES testing to account for such variables as equipment error, torque switch repeatability, and rate of loading effects consistent with NRC Generic Letter 89-10. The resultant "as left" thrust value was measured to be 17,587 pounds. The inspector determined the adjustments to the torque switch and resultant thrust to be appropriate to assure valve operability.

8.2 Reactor Vessel Water Level Instrumentation Spiking

In recent years, the licensee has experienced reactor vessel water level instrumentation spiking during reactor depressurization following plant shutdowns. Typically, the level instrumentation spiking has been observed to begin at less than 600 psig reactor pressure and has been much more prominent on the reactor vessel "B" reference leg instrumentation. The spiking has been of sufficient magnitude to cause several automatic primary containment isolation system (PCIS) Group I isolations. Corrective actions to previous level instrument spiking included improved sensing line backfill procedures and implementation of a modification which increased the reactor vessel to condensing chamber equalizing [illegible] diameter from one inch to two inches.

Notwithstanding these corrective actions, level instrument spiking and three automatic Group I PCIS actuations occurred during the March 26 and 27 post shutdown reactor depressurization. The second of the three isolations was the result of an actual high vessel water level and is documented in Section 2.3.

As during previous events, the March 26 spiking was initially observed at approximately 450 psig reactor pressure and was initially limited to "B" reference leg instrumentation which did not result in PCIS actuation. The spikes were typically of 30 seconds duration, were similar to a square wave recorder trace, and were of approximately positive four inches in amplitude. Level spiking on "A" reference leg instrumentation was initially observed at 65 psig reactor pressure, but the spikes were typically of positive one to two inches in amplitude.

Both the first and third automatic Group I isolations were initiated from instrument spiking on the "B" reference leg. The first isolation occurred on March 26 at 8:59 pm, with the reactor shutdown and at approximately 55 psig pressure when the "B" reference leg instrumentation experienced a spike of positive nineteen inches from +29 to +48 inches. Instrumentation on the "A" reference leg remained unchanged through this occurrence. The third isolation occurred on March 27 at 5:45 am, with the reactor at 10 psig pressure and with the shutdown cooling system being placed in service. Instrumentation on the "B" reference leg spiked positive eighteen inches from +29 inches to +47 inches causing the PCIS actuation. Instrumentation on the "A" reference leg spiked positive fourteen inches from +29 inches to +43 inches but remained below the PCIS actuation setpoint. All components responded to each Group I isolation signal as designed.

The licensee formed a root cause analysis team (RCAT) to further investigate the continuing level instrumentation spiking. In addition to licensee personnel from operations, system engineering, and nuclear engineering, the team received technical expertise from General Electric Company and Yankee Atomic Electric Company instrumentation specialists. The team determined the root cause of the level instrumentation spiking to be unsatisfactory thermodynamic performance of the "B" reference leg condensing chamber and associated steam drain line. The team concluded that, during reactor depressurization the temperature of the condensing chamber and drain line metal surfaces exceeds the saturation temperature of the reactor coolant, causing condensate in the drain line to vaporize and flow rapidly into the reactor vessel. This action would cause more vapor to vacate the condensing chamber, creating a reduced pressure condition within the chamber. The reduced chamber pressure would be sensed by the level instrumentation as a high vessel level spike.

The team also identified the buildup of non-condensible gasses in the condensing chamber as a potential contributor to these events. It was believed that, as non-condensible gasses accumulate in the chamber during plant operations, the condensation rate is decreased which in turn reduces the chamber and drain line heat transfer capability.

In order to improve heat transfer from the condensing chamber and drain line to the drywell atmosphere which would serve to reduce surface metal temperatures, the licensee removed the "B" reference leg condensing chamber and drain line insulation via a temporary modification (TM 92-13). Additionally, more temporary temperature instrumentation was installed (via TM 91-44) on both the "A" and "B" reference leg condensing chambers and associated drain lines in order to monitor component thermodynamic performance during power operations and during reactor shutdowns. This instrumentation should also provide capability to trend potential buildup of non-condensible gasses in the condensing chambers.

8.2.1 Engineered Safety Features Actuation During Troubleshooting

The RCAT also identified trapped air in the sensing lines as a potential contributing factor to the instrument spiking. Although the licensee previously implemented procedures to improve sensing line backfill, the lines had never been verified to be free of trapped air. Therefore, in order to evaluate this potential, the licensee developed Temporary Procedure, TP 92-20, "Reactor Level Instrument Line Test and Investigation on Rack 2206 (Constant Level, Decreasing Pressure)." The procedure was intended to be a test which simulated a reactor shutdown by decreasing a static pressure applied to the level instrumentation.

On April 3, TP 92-20 was initiated. Procedure step 10.?.2 directed the removal of analog trip system (ATS) master trip units for reactor level and pressure which input into the emergency core cooling system (ECCS) logic. However, the sequence established by the procedure created a configuration in which "B" logic circuitry slave trip units LS-263-72D-1 and LS-263-72B-1 concurrently received low-low reactor vessel water level trip signals when their respective master trip units were removed. The concurrent signals satisfied ECCS initiation logic which resulted in the automatic start of both emergency diesel generators, automatic start of the "A" train of RHR with associated valve repositioning, repositioning of associated valves in the "B" train of RHR (which was in the shutdown cooling mode of operation), and the automatic opening of HPCI steam supply and injection valves. Because the reactor was shutdown, the HPCI system did not initiate. Additionally, because the LPCI cross-tie line was isolated during the outage and because the LPCI loop select logic was selected to the "B" loop, the "A" train RHR pumps operated in minimum flow recirculation. All affected equipment responded to the ECCS initiation signal as designed.

The test was immediately terminated, the master trip units were reinstalled and the ATS logic was reset, and normal safety system status was restored. Technically, the event had minimal safety significance. Decay heat removal was maintained throughout the event and all systems and components performed as designed.

The licensee promptly conducted a thorough review of this event. Proper system responses were verified and a causal analysis was initiated. The cause of this event was determined to be inadequate procedure development and review. As a result, Temporary Procedure 92-20 did not establish appropriate actions to preclude the actuation. Additionally, subsequent reviews of the procedure by a procedure validator and the Onsite Review Committee failed to identify the procedure deficiency. Specifically, the reviewers did not identify that removal of master trip units with the respective slave trip units that have low reactor water level or low reactor pressure functions in service, would cause the associated trip relays to be energized.

The individuals involved in the development and issuance of TP 92-20 were counselled to ensure their responsibilities were understood. Subsequently, the licensee retired the faulted procedure and generated a new procedure (TP 92-22) to test for the presence of air in the sensing lines. This procedure required the individual slave trip unit to be removed before its associated master trip unit. Procedure TP 92-22 was performed without further incident on April 5-7.

Technical Specification (TS) 6.8.A requires that written procedures shall be established and implemented that meet or exceed the requirements of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USNRC Regulatory Guide 1.33. Additionally, TS 6.8.B requires that such procedures be reviewed by the ORC and approved by the responsible department manager.

Contrary to these requirements, TP 92-20 was established and implemented with deficiencies that caused a partial ECCS actuation. Further, the ORC review failed to question the deficiencies. Notwithstanding, the licensee effectively identified the cause of this event, initiated prompt corrective actions, and properly reported the event in Licensee Event Report 92-05 (Section 7.2.5). Therefore, the licensee-identified violation is not being cited because the criteria specified in Section VII.B of the NRC's Enforcement Policy were satisfied.

8.2.2 Temporary Waiver of Compliance and Post Modification Testing

The RCAT concluded the most probable cause for the level instrumentation spiking was unsatisfactory "B" reference leg condensing chamber thermodynamic performance. Additionally, the team concluded with reasonable assurance that removal of the condensing chamber and drain line insulation would be the appropriate corrective action to the spiking. By letter dated April 7, the licensee requested a Temporary Waiver of Compliance (TWOC) from TS limiting conditions for operations requirements associated with the affected reactor vessel water level instrumentation. The purpose of the waiver request was to permit reactor startup to not more than 15% of rated power to conduct post modification (i.e. insulation removal) testing of the "B" reference leg instrumentation. The licensee concluded the waiver request was necessary to comply with the post modification testing requirements as stated in the Boston Edison Quality Assurance Manual. The waiver request included a detailed bounding safety evaluation. After comprehensive staff review, a Regional Waiver of Compliance was granted consistent with NRC letter dated April 8.

On April 9 at 8:59 am, the licensee commenced reactor restart. Reactor power was increased to approximately 12% where it was maintained for approximately ten hours to allow equilibrium temperatures to be achieved. On April 10 at 6:18 pm, reactor shutdown was initiated and the reactor entered cold shutdown on April 11 at 3:35 am. Reactor vessel level and condensing chamber performance data was recorded throughout the reactor power evolution in accordance with post modification test TP 92-21. The results of the test evolution identified all instrumentation level spikes to be one inch or less in magnitude. Based on these results, the licensee declared the affected instrumentation operable and prepared the station for reactor restart to full power operations.

The post modification testing was extremely well controlled. Control room operators maintained excellent awareness of all test related activities, including outstanding control of all reactivity manipulations. Test coordinators ensured complete data acquisition and analysis. Although the test was effective in evaluating most aspects of condensing chamber thermodynamic performance, a test limitation was the inability to establish potential effects of the buildup of non-condensible gasses. Extended level instrument performance with respect to potential non-condensible gas buildup effects is identified as an unresolved item (UNR 50-293/92-04-02).

The licensee investigation of the continuing level instrumentation spiking was very well controlled with the noted exception of the unplanned partial ECCS actuation. Licensee management provided the RCAT with necessary support and technical expertise. The RCAT evaluations were comprehensive and reflected safety conscious perspectives. The TWOC and associated safety evaluation were effectively supported by design and licensing bases documentation. Post modification testing was performed in a deliberate and conservative manner.

9.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (30702)

9.1 Routine Meetings

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

9.2 Management Meetings

On April 7, a conference call was conducted between representatives of NRC: Region I, NRR, and the licensee to discuss operability of certain reactor vessel water level instrumentation and a related Temperary Waiver of Compliance request. This subject is discussed further in Section 8.2.2 of this report.

9.3 Other NRC Activities

On March 18, Mr. Thomas Martin, the Regional Administrator, NRC:Region I toured PNPS and met with licensee management to discuss current licensee performance.

On May 1, NRC Chairman Ivan Selin, Executive Director for Operations James Taylor, and Region I Administrator Thomas Martin toured PNPS and met with Boston Edison Company corporate officers to discuss current performance and future licensee initiatives. A press conference was conducted at the conclusion of the site visit.

On May 3, Mr. John Rogge, NRC Region I Chief, Reactor Projects Section 3A responsible for inspection program management at PNPS was reassigned to become Chief, Reactor Projects Section 4B. Mr. Eugene Kelly, previously Region I Chief, Reactor Projects Section 4A, has succeeded Mr. J. Rogge.