U.S. NUCLEAR REGULATORY COMMISSION
REGION I

License No.:	DPR-35
Report No.:	99-04
Docket No.:	50-293
Licensee:	Entergy Nuclear, Inc. Pilgrim Nuclear Power Station 600 Rocky Hill Road Plymouth, Massachusetts 02360
Facility:	Pilgrim Nuclear Power Station
Inspection Period:	June 10, 1999 through July 25, 1999
Inspectors:	R. Laura, Senior Resident Inspector R. Arrighi, Resident Inspector J. D'Antonio, Operations Engineer
Approved by:	C. Anderson, Chief Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Pilgrim Nuclear Power Station
NRC Inspection Report 50-293/99-04

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers resident inspection for the period of June 10, 1999, through July 25, 1999; in addition, it includes the results of an announced inspection by an operations engineer to review the corrective actions implemented to address the NRC maintenance rule inspection.

Operations

- The pre-evolution briefing and operator performance for starting of a reactor recirculation pump at low power was very good. All personnel involved were trained on the evolution using the simulator prior to the starting of the recirculation pump. (Section O2.1)

- Overall, operator performance during reactor startup and power ascension activities was good. Operators responded effectively to two reactor vessel level transients; prompt response averted a plant transient or reactor scram when the feedwater regulating valve failed open. (Section O4.1)

- The failure of the reactor operators to verify plant conditions while placing the residual heat removal system in the low pressure coolant injection mode of operation resulted in a reduction of ten inches is reactor vessel level. This level 4 procedure violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as PR 99.9380. (NCV 50-293/99-04-01). Interim corrective actions to improve configuration control problems were effective in preventing further problems during the reactor startup. (Section O4.1)

Maintenance

- The identification of a degraded bleeder resistor for the "A" recirculation pump was identified by an alert maintenance supervisor on tour. (Section O2.1)

- Good pre-job briefs and procedure adherence was displayed during maintenance and surveillance activities. Thorough planning through the use of just-in-time training resulted in the licensee establishing plant conditions that would lesson the severity of a feedwater transient during troubleshooting of the feedwater regulating valve. Good oversight was noted by the quality assurance staff during maintenance activities. (Section M1.1)

- Problems were experienced with MSIVs in the latter part of RFO12 due to several different work quality issues. Examples included the existence of a high spot on the seating surface, pinched O-ring, flow control valve installed backwards, stem plate retaining bolt not fully seated, and loose union connecting an accumulator to the

pneumatic supply. Additional worker radiation exposure was needed to perform maintenance rework to correct these conditions. Inadequate oversight and experience of contract workers contributed to these issues which were detected and corrected prior to restart from the outage. (Section M7.1)

- The corrective actions implemented in response to the NRC maintenance rule team inspection were good. Quality assurance performed a detailed audit in the maintenance rule area and identified opportunities for further improvement. This audit was considered a strength. (Sections M8.2, M8.3, M8.4)

Engineering

- The water hammer event in the feed water system resulted from the combination of leakage through the long path recycle line and the new leak tight design of the new feed water system regulating valves. The licensee did not fully consider the possible impact of the degraded valve leakage with the installation of the new leak free feed water regulating valves. (Section O4.1)

- Systems engineering personnel increased the margin for the close stroke time of AO-220-45 by developing and installing a modification to allow quicker porting of the actuator air. The modification significantly reduced the stroke close time below the 10 second limit listed in the UFSAR. (Section E2.1)

- A deficiency with the "2C" MSIV actuator speed control valve was thoroughly evaluated and corrected. The problem scope review identified corrective actions for three other MSIVs which were corrected prior to restart from RFO12. Also, good communication with the vendor and between licensee engineering and maintenance personnel was observed. (Section E2.2)

- The inspector identified an EDG TMOD and related safety evaluation did not evaluate the degradation of the fire suppression system. This occurred due to inadequate communications within the engineering department as well as an improper field walk down. The failure to evaluate the impact of the TMOD on the fire sensor is considered a violation of NRC design requirements. This level 4 violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as PR 99.9405. (NCV 50-293/99-04-02) (Section E4.1)

Plant Support

- The dose rates in the RHR quadrants increased after the refueling outage to levels established prior to the chemical decontamination effort performed in the fall of 1998. (Section O1.1)

- Security assessment aids were operating properly in surveying the site grounds. A past NRC violation was closed. (Section E8)

TABLE OF CONTENTS

Table of Contents (cont'd)

ATTACHMENTS

REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station (PNPS) began the period shutdown and in refueling outage 12 (RFO12), which commenced May 7, 1999. Operators brought the reactor critical at 08:31 on June 28 and the mode switch was taken to "run" on July 1. After replacement of the main electrical transformer on July 7, the unit was placed on-line. Operators continued to increase power and performed the required testing at the appropriate power levels. On July 15, at 10:38, the reactor was taken to 100 percent power. Minor power reductions were performed for feedwater regulating valve troubleshooting and final rod pattern adjustment. The reactor was at 100 percent power at the end of the inspection period.

The transfer of the Pilgrim operating license from Boston Edison Company to Entergy Nuclear Generation Company became effective on July 13, 1999, after completion of the cycle 12 refueling outage. Upon the license transfer, Mr. Robert M. Bellamy was appointed the Site Vice President (the senior site representative).

I. OPERATIONS

O1 Conduct of Operations[1]

O1.1 General Comments (71707)

Using inspection Procedure 71707, the inspector conducted frequent reviews of ongoing plant operations, including operator evaluations in the control room; walk-down of the main control boards; tours of radiological controlled areas; and observations of management planning meetings. The inspector observed that proper control room staffing was maintained. Shift briefings and turnovers were well conducted with good discussion on compensatory measures and degraded equipment. Management meetings were attended by all departments and discussions included present plant conditions, identified equipment problems and recent industry operating experience.

During tours of reactor plant spaces, the inspector noted improvement in the cleanliness of reactor plant spaces and several material condition improvements to the secondary plant with the installation of newly designed feedwater regulating valves and the replacement of 18 steam dump and level control valves. The inspectors also noted that the dose rates in the RHR quadrant rooms have increased. The dose rates in these areas exceed the levels established prior to the chemical decontamination effort performed in the fall of 1998.

Anomalies identified on tour were discussed with the nuclear watch engineer. As an effort to improve control room decorum and formality, operators were provided uniforms and the Operations and Plant Manager developed written guidance on operations standards. Uniforms had been previously worn only by the senior reactor operators and

---

[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

operator standards were communicated verbally. The inspector noted that three-way communications have improved since the implementation of these changes.

O2 Operational Status of Facilities and Equipment

O2.1 Reactor Recirculation Field Breaker

a. Inspection Scope (71707)

The inspector reviewed the circumstances regarding damage to the "A" reactor recirculation bleeder resistor and observed the subsequent startup of the reactor recirculation pump at low power.

b. Observations and Findings

On June 30, 1999, with the plant at one percent reactor power, an electrical maintenance supervisor noted a burning odor in the reactor recirculation motor generator set room. Investigation found the bleeder resistor for the "A" recirculation pump glowing red hot. Operators secured the "A" recirculation pump from service, de-energized the breaker, and declared the recirculation pump inoperable. Operators properly entered a 24 hour reactor shutdown action in accordance with step 3E of the Facility Operating License.

Licensee investigation revealed that the spacer on the GE AK25 field breaker was missing. When the field breaker is racked in, the spacer pushes down on a plunger/disc which removes the center bus bar from the circuit. Without the spacer, the bleeder resister (used for arc suppression) remains in the circuit when the breaker is closed due to the plunger/disc contact with the center bus bars. This configuration should not affect the operability of the recirculation motor generator set. In the event the field discharge circuit did not function (i.e. burned/melted wire) this could present a personnel safety issue in that any stray field current would not be dissipated.

Inspection of the "B" train reactor recirculation breaker revealed no problems. The licensee indicated that there are no other breakers in the plant which have this configuration. This breaker configuration is applicable only for GE AK25 DC field breakers. Preliminary review revealed that when this breaker was procured in 1993, as a spare, it did not have the spacer.

The inspector monitored the startup of the "A" recirculation pump at low power. This was a first time evolution for Pilgrim to start a recirculation pump at low power with reactor pressure at approximately 320 psig. The operating crew was sent to just-in-time training at the simulator to practice the evolution to understand the expected plant response, and to ensure they had good control over the rate of positive reactivity insertion and ranging of the intermediate range monitors. The concern with starting a recirculation pump at low power level is the addition of positive reactivity and the effect on reactor power and pressure. The pre-evolution brief and operator performance was excellent. The evolution was well coordinated and controlled. The brief covered specific personnel responsibilities, abort criteria, expected alarms and past operational events from other

utilities. Dedicated operators were appointed to the various stations to monitor plant response. Also, the licensee implemented peer checking to ensure that the evolution was properly performed. The "A" recirculation pump was started with no problems and the licensee exited from the 24 hour shutdown action requirement.

c. Conclusions

The identification of the damage to the bleeder resistor for the "A" recirculation pump was attributed to an alert maintenance supervisor on tour.

The pre-evolution briefing and operator performance for starting of a reactor recirculation pump at low power was very good. All personnel involved were trained on the evolution using the simulator prior to the starting of the recirculation pump.

O4 Operator Knowledge and Performance

O4.1 Startup From Refueling

a. Inspection Scope (71707)

The inspector observed portions of the reactor startup and power ascension activities to monitor operations department and overall equipment performance following the cycle 12 refueling outage (RFO12).

b. Observations and Findings

The inspector attended the June 27, 1999, brief and monitored operator performance for procedure 2.1.1, "Startup From Shutdown." The brief was determined to be adequate. The brief covered major evolutions that were scheduled to be performed with minimal detail; precautions and potential plant problems were not reviewed. Licensed operators questioned the nuclear operating supervisor about potential problems (i.e., stuck rod) and were informed that these would be covered just prior to reactor startup. Discussions with the Assistant Operations Manager revealed that operating crews had attended just-in-time training on the simulator.

The inspector verified that operators followed procedure 2.1.1 and independently verified selected plant conditions were established as required by procedural steps. In preparation for reactor startup, operators secured from the shutdown cooling (SDC) mode of residual heat removal in accordance with procedure 2.2.19 and aligned the system for automatic initiation (LPCI mode). While clearing the SDC tagout, operators opened minimum flow valve MO-1001-18A which resulted in a lowering of reactor pressure vessel (RPV) level of approximately ten inches. This condition was not noted until the RPV low level alarm was received. Upon receipt of the alarm, operators immediately closed valve MO-1001-18A and restored vessel level to normal. The SDC suction and discharge valves were open, thus resulting in flow from the vessel to the torus.

Preliminary review of the event revealed that the control room operator who was performing the procedure steps to secure from SDC lineup stopped prior to aligning the SDC suction and discharge valves due to control room tags to be removed by the SDC tagout. The operator turned over the status of RHR system lineup to the nuclear operation supervisor (NOS) and momentarily left the control room for a break. The nuclear plant operator clearing the SDC tagout entered the control room and the NOS directed the second on shift reactor operator to complete the control room tagout. The operator assumed that since the NOS directed him to remove the control room tags the system lineup was correct and he did not use self-checking techniques prior to performing the actions.

As a result of this event and prior configuration control problems (tagging errors; refer to NRC IR 50-293/99-03 section O4.1), the licensee implemented peer checks for control room and field manipulations where practicable. Operators brought the reactor critical on June 28 without further operator performance errors. The failure to secure from the SDC mode of operation per procedure 2.2.19 is a violation of technical specification 6.8.A. This Severity Level IV violation is being treated as a Non-Cited Violation consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as PR 99.9380. (**NCV 50-293/99-04-01**)

Several equipment/system issues emerged during the start-up which adversely affected operational activities. While at one percent reactor power, operators declared the "A" reactor recirculation pump inoperable. The bleeder resistor for the "A" recirculation pump was found to be damaged as a result of a field breaker problem (refer to section O2.1). The breaker was replaced and power ascension recommenced.

A second equipment issue that affected power ascension involved damage to the "A" feedwater regulating valve (FRV). In preparation of placing the FRVs in service at 20 percent power, the first point heater downstream block valve MO-3479 was jogged open. Upon opening of the valve, operators heard a rumbling and loud bang (water hammer) and noted the "A" FRV went full open. Reactor vessel level rose and the RPV high level alarm annunciated. Operators immediately shut valve MO-3479 and restored RPV level using the startup feedwater regulating valve. Prompt operator response averted a potential plant transient. Discussions with the Operations Department Manager revealed that prior to placing the FRV in service the operating crew was briefed on feedwater system malfunctions.

A walk down of the area by the licensee revealed no damage to the feedwater piping, hangers, or supports. The licensee postulated that the water hammer (mechanical shock) resulted in the failure of the current-to-air controller for the "A" FRV resulting in the valve to fail open. Operators had filled and vented the piping between the first point feedwater heater "A" and the FRVs per the startup procedure; however, the licensee postulated that leakage from the long path recycle line to the condenser resulted in depressurization and partial voiding of the piping.

The inspector noted that new FRVs were installed during the outage. The FRVs were designed leak tight as opposed to the old design which had designed leakage. During

the refueling outage the feedwater long path recycle line valves were scheduled to be replaced due to known valve seat leakage. However, due to parts availability and outage duration, the condition was not corrected. The licensee concluded that the valve leakage by the long path recycle line valves would not affect safe plant operation; only plant efficiency. The licensee didn't fully consider the possible impact of the degraded valve (leakage) with the installation of the new (leak free) feed water regulating valves. Problem report PR99.9392 was written to document this event.

c. Conclusions

Overall, operator performance during reactor startup and power ascension activities was good. Operators responded effectively to two reactor vessel level transients; prompt response averted a plant transient or reactor scram when the feedwater regulating valve failed open. This event resulted from leakage through the long path recycle line and the leak tight design of the new feed water system regulating valves. The licensee did not fully consider the possible impact of the degraded valve leakage with the installation of the new leak free feed water regulating valves.

The failure of the reactor operators to verify plant conditions while placing the residual heat removal system in the low pressure coolant injection mode of operation resulted in a reduction of ten inches is reactor vessel level. This level 4 procedure violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as PR 99.9380. **(NCV 50-293/99-04-01)**. Interim corrective actions to improve configuration control problems were effective in preventing further problems during the reactor startup.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 General Maintenance and Surveillance

a. Inspection Scope (61726/62707)

The inspector observed portions of selected surveillance and maintenance activities to verify that the applicable procedures and technical specifications were satisfied, proper system restoration, and that the post work testing was adequate.

b. Observations and Findings

The inspector observed all or portions of the following activities:

MR E9800105, Trouble Shooting Feedwater Regulating Valve (FRV) Oscillation

The inspector attended the prejob brief for the instrumentation of the FRVs and observed the installation of test equipment in the control room. The FRV was instrumented to gather data on valve response and feedwater level control system.

Feedwater flow oscillations increased after startup from the refueling outage (RFO). The FRVs had been modified during the RFO as part of plant design change 98-29 to allow a greater valve stroke and eliminate oscillation problems.

The inspector concluded that the preplanning and prejob brief was thorough. Operators were sent to just-in-time training at the simulator to practice feedwater malfunctions (failed open and closed FRV). As a result of this training the licensee discovered that the placement of one FRV in manual would reduce the severity of a feedwater transient. Instrument and control technicians also participated in the simulator training and rehearsed installing the test jacks. No problems were experienced during the field observation.

SP 8.5.4.1, High Pressure Coolant Injection Operability Determination

The inspector verified that the licensee appropriately implemented technical specification (TS) surveillance requirements. Good procedure adherence was displayed by the maintenance craft. The surveillance procedure acceptance criteria was revised to demonstrate the pump could deliver the required flow at the upper analytical setpoint of the safety relief valves.

During the conduct of RFO12, the inspector noted that on several occasions the quality assurance staff raised concerns on projects that resulted in work stoppage. This was viewed as a positive action to identify potential problems prior to their identification during a consequential event.

c. Conclusions

Good pre-job briefs and procedure adherence was displayed during maintenance and surveillance activities. Thorough planning through the use of just-in-time training resulted in the licensee establishing plant conditions that would lesson the severity of a feedwater transient during troubleshooting of the feedwater regulating valve. Good oversight was noted by the quality assurance staff during maintenance activities.

M7 Quality Assurance in Maintenance Activities

M7.1 Main Steam Isolation Valve (MSIV) Work Problems

a. Inspection Scope

The inspector reviewed the issues involving the MSIVs that surfaced late in RFO12 which had the potential to be classified as maintenance rework. The inspector reviewed maintenance work package notes and interviewed the system engineer and work control personnel. The majority of work performed on the MSIVs was done by contract workers under the control of the licensee. The problems mostly occurred with work performed on the inboard MSIVs which are located inside the drywell. As a result, the maintenance rework resulted in additional radiation worker exposure.

b. Observations and Findings

The internal seating surfaces of the "C" steam line inside containment MSIV (i.e., AO-203-1C) were refurbished under MR 19702038 including machining of the seating surface. After valve re-assembly, valve AO-203-1C was tested with unsatisfactory results. A failed LLRT indicated a problem with the seating surfaces. The valve was disassembled and licensee inspection identified an .008 inch high spot on a small portion of the seating surface. A different machining tool was utilized and the seat was machined to within specifications. After AO-203-1C was re-assembled, a LLRT was reperformed with successful results.

The inspector questioned the licensee on why the high spot on the seat was not identified during the initial work. The system engineer indicated that the high spot was either missed during the final inspection or the original tooling used to machine the seating surface was slightly defective due to worn bearings. The engineer further indicated that discussions were ongoing with the vendor to inspect the tooling to identify the root cause of the high spot on the seating surface. The inspector noted that during the initial work on the seating surface, the licensee did not independently inspect the seating surface for adequacy. The inspector noted that no problem report or rework evaluation was initiated by the licensee. However, the system engineer and refueling outage manager indicated that these issues would be reviewed in detail during the post refueling outage lessons learned review.

A second MSIV maintenance problem became evident on valve AO-220-1A during the limit switch and functional testing per procedure 8.M.1-15. During valve stroking , the valve stem moved only 8.25 inches which is less than the low end of the acceptance criteria of 9 inches of travel. Maintenance troubleshooting identified several maintenance quality work issues from work performed by contractor personnel. The stem plate retaining bolt was found not fully engaged. This was corrected but the valve closed in erratic short strokes due to binding. Further troubleshooting revealed that the actuator hydraulic fluid flow control valve was found installed backwards. After this was corrected, the valve stroked satisfactorily. The licensee issued problem report PR 99.9345 to document these issues and initiated a rework evaluation.

A third MSIV problem involved valve AO-203-1B which had oil leakage from the bottom of the actuator cylinder. This portion of the actuator was not worked during RFO12 so the corrective maintenance needed was not initially considered rework. However, several iterations of corrective maintenance were required to fix the oil leak. Problems were experienced in the bottom pipe at the actuator not being fully tightened and also the replacement oil seal was pinched during assembly. The licensee initiated PR 99.9344 to document these issues and to perform a rework evaluation.

A fourth MSIV problem was disclosed during the pressure drop test of accumulator T-220 which services valve AO-203-1C. The accumulator failed the air drop pressure test. Maintenance troubleshooting identified a loose union flexible coupling connecting the accumulator to the pneumatic supply. The accumulator had previously passed its pressure drop test during RFO11, thus the test failure was attributed to a problem with

reconnection of the accumulator during work performed during RFO12. The licensee initiated PR 99.9358 to document and evaluate this deficiency. No rework evaluation was initiated.

The inspector discussed these work quality issues with the maintenance department and quality assurance managers. The quality assurance manager indicated that a broader review of all rework activities was in progress by the quality assurance staff. Likewise, the maintenance manager indicted that a lessons learned review was planned for the work completed during RFO12. The inspector concluded that several iterations occurred during MSIV work that were related to quality issues such as inadequate oversight of contract workers.

c. Conclusions

Problems were experienced with MSIVs in the latter part of RFO12 due to several different work quality issues. Examples included the existence of a high spot on the seating surface, pinched O-ring, flow control valve installed backwards, stem plate retaining bolt not fully seated, and loose union connecting an accumulator to the pneumatic supply. Additional worker radiation exposure was needed to perform maintenance rework to correct these conditions. Inadequate oversight and experience of contract workers contributed to these issues which were detected and corrected prior to restart from the outage.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) LER 50-293-99-01: Environmental Enclosures of Motor Control Centers (MCCs) Outside Design Basis

This LER documented that the pressure relieving devices that are part of the environmental enclosures for three safety-related MCCs were found to be obstructed or retarded from opening. The devices are designed to open if an internal pressure of 0.5 psi occurs as a result of a design basis tornado. Problem report 99.9126 was written to document and evaluate this condition.

The inspector conducted an on-site review of the LER and reviewed the licensee's corrective actions and determined that they were appropriate. The inspector verified that the obstruction has been removed from the MCCs. No violation of NRC requirements were identified. This LER is **closed.**

M8.2 (Closed) EA 98-277/01014: Failure to establish unavailability measures for the anticipated transient without scram (ATWS) function of the control rod drive system.

The inspector reviewed the licensee's corrective actions for three violations identified in Maintenance Rule Baseline Team Inspection report 50-293/98-04. These corrective actions were described in the licensee's response to the notice of violation in a letter dated August 28, 1998.

The inspector reviewed the design basis document for Control Rod Drive and Hydraulics, System 3. The ATWS mitigation function had been incorporated with an unavailability criterion based on estimated time to repair allowed functional failures plus preventive maintenance time.

M8.3 (Closed) EA 98-277/02014: Failure to include in the scope of the maintenance rule the following two structures, systems, or components: Heating, Ventilation, and Air Conditioning (HVAC) for 480V switchgear enclosures in the reactor building, and the firewater system function of providing a backup screenwash supply.

The inspector reviewed the design basis documents for HVAC, System 24 and Fire Protection, System 33. Both documents had been modified to include the previously omitted items with adequate justification for the performance criteria provided.

M8.4 (Closed) EA 98-277/03014: Failure to complete the periodic evaluation of maintenance rule activities required by 10CFR50.65(a)(3) in a timely manner.

The inspector reviewed the last periodic evaluation covering the period July 10, 1996 through May 5, 1998. Based on this limited review, this document adequately addressed the elements identified in 10 CFR 50.65 and NUMARC 93-01 Rev 2 as endorsed by Reg Guide 1.160. Pilgrim Nuclear Power Station Procedure No. 16.03 "10CFR50.65 NRC Maintenance Rule" had been modified to clarify timeliness expectations.

The facility had issued problem report (PR) 98.0919 to address these violations; corrective action for this PR was a comprehensive self assessment of sixty-six systems for adequacy of scoping and performance criteria. This self assessment resulted in fifty-five additional PRs which the facility prioritized for resolution. The facility had also completed a QA audit of the resolution of these PRs and of findings from earlier audits and surveillances. This was a thorough and critical audit and identified continuing instances where the facility was not meeting their own timeliness goals for resolution of self assessment findings or development and updating of (a)(1) corrective action plans. PRs had been generated in response to these audit findings. The inspector interviewed the individual performing the root cause evaluation for the continuing timeliness and considered his preliminary findings to be reasonable. At the time of this inspection, five systems had been placed in (a)(1) status as a result of self assessment findings. None were due to equipment failures; one system was meteorological tower instrumentation which had been added to scope and had not had criteria developed, the other four were systems for which self assessment findings concerning scoping of some functions had not yet been resolved.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 (Closed) IFI 50-293/98-11-04: AO-220-45 Stroke Time

The outboard reactor water sample line containment isolation valve AO-220-45 failed its quarterly stroke-close surveillance test. The UFSAR specifies that valve AO-220-45 must be capable of stroking closed in ten seconds or less. The valve was declared inoperable, interim corrective actions were taken and final corrective actions were planned during RFO12. During this inspection period, the air-operator for AO-220-45 was removed and preventive maintenance was performed. The valve bonnet was repacked with new packing. Subsequently, during the post maintenance test, the valve stroked closed in less than the 10 seconds.

Although the valve stroked within specification, systems engineering determined that less margin for the stroke-close time existed than was expected. As a result, engineering developed and implemented a modification to allow the actuator air to bleed off faster allowing the valve to stroke close quicker. Subsequently, during the post modification test, the valve stroke-closed in significantly less than 10 seconds. System engineering also evaluated other air-operated valves to ensure that adverse trends did not exist in stroke times. No other significant problems were identified by the licensee. The inspector determined that the effort to develop and modify valve AO-220-45 to increase the closing time margin was a positive effort to avoid future testing problems. This item is **closed**.

E2.2 Main steam Isolation Valve (MSIV) Speed Control Valve Design Issue

a. Inspection Scope

A review was performed of the cause and corrective actions for the identification of a degraded speed control valve on the actuator for the 2C MSIV (i.e., AO-220-2C). The inspector visually examined the degraded parts, interviewed the system engineer and reviewed the relevant part specifications. Related problem report PR 99.9359 was also reviewed.

b. Observations and Findings

While troubleshooting erratic operation of valve AO-220-2C during RFO12, the maintenance staff identified that three internal parts of the MSIV actuator speed control valve were missing. Specifically, a plastic cage was missing, a spring was found in the oil cylinder and a check ball was located in the oil piping. The system engineer contacted the vendor of the speed control valve for technical assistance. The vendor determined that the degraded speed control valve was intended for temperature applications only up to bulk temperatures of 200 degrees Fahrenheit. The system engineer indicated that actual temperatures exceeded 200 degrees Fahrenheit. Hence,

the vendor and licensee determined that the non-metallic cage, which houses the spring and check ball, broke down in the high temperature hydraulic oil and disintegrated.

Engineering personnel prepared and implemented Field Revision Notice (FRN) 99-01-67 to replace the degraded speed control valve with a different model that contained metallic parts and was designed for higher temperatures. As part of the problem scope review, the licensee determined that three other MSIVs had the same low temperature model and required replacement. Two of these were installed during RFO12 and the other two in 1989. The remaining four MSIVs were original equipment and had speed control valves with metallic parts. The inspector determined that the licensee conducted an effective problem scope review and the corrective actions to replace the four low temperature speed control valves was completed prior to restart from RFO12. Also, Effective teamwork and communications was observed between engineering and maintenance personnel during resolution of this issue.

c. Conclusions

A deficiency with the "2C" MSIV actuator speed control valve was thoroughly evaluated and corrected. The problem scope review identified corrective actions for three other MSIVs which were completed prior to restart from RFO12. Also, good communication with the vendor and between licensee engineering and maintenance personnel was observed.

E4 Engineering Staff Knowledge and Performance

E4.1 Improper Alterations to EDG Temporary Modification (TMOD) on High Ambient Air Temperature

a. Inspection Scope

On July 5, 1999, the licensee declared both EDGs inoperable when outside ambient air temperature exceeded 88 degrees Fahrenheit. Operations personnel made the requisite formal NRC one hour ENS notification. The EDGs design temperature for ambient air is 88 degrees; however, efforts are underway to implement permanent modifications to increase the design temperature to 95 degrees. In the interim, until the permanent modifications were made, the licensee implemented TMOD 99-42 to restore EDG operability up to an outside ambient air temperature of 95 degrees. The inspector reviewed TMOD 99-42 and inspected the field implementation.

b. Observations and Findings

TMOD 99-42 implemented three changes to the EDG. The air bypass flow was blocked around the perimeter of the EDG radiator housing which serves to improved cooling of the radiator fluid. Secondly, the engine side air plenum doors were blocked open to improve air flow through the engine room and radiator. Lastly, the doors of the local electrical control panels located inside the EDG room were pinned open to allow for better cooling of internal electrical control components. Collectively, these changes

assisted in maintaining the EDG process operating parameters within established vendor limits. The licensee completed safety evaluations 3264, dated July 2, 1999, and 3265, dated July 3, 1999, to support the changes made implemented by the TMOD. The Safety evaluations concluded that no unreviewed safety questions existed. The licensee implemented TMOD 99-42 for both the EDGs.

The inspector verified the adequacy of the implementation of TMOD 99-42 in the field. The inspector identified a problem related to the east air plenum door for the "A" EDG. There was an ultra-violet fire sensor (i.e., B1-2) which was partially blocked when the door was pinned open. The inspector discussed this concern with fire protection personnel who indicated that the fire sensor was designed to view an arc of 120 degrees. The inspector estimated that with the door blocked open, the fire sensor could only sense approximately 20 degrees. The fire protection engineer repositioned the door in a manner that allowed the fire sensor to function normally. The inspector reviewed TMOD 99-42 and determined that the TMOD and supporting safety evaluations did not adequately evaluate the effect of pinning the door open on the operability of the fire sensor. The licensee initiated a problem report to document, evaluate and correct this problem.

A few weeks later the inspector again identified that the same door had been repositioned in a manner that blocked the fire sensor. The inspector informed the NOS who initiated corrective action. The NOS directed that the door be removed rather than pinning the door in the mid-position. The inspector had no further concerns on this issue.

The inspector determined that the TMOD and safety evaluation were deficient since they did not evaluate the impact on the fire sensor. However, the fire engineer indicated that due to other fire sensors in the room, in all likelihood, the fire system would have actuated the water suppression system in the event of an actual fire. The failure to evaluate the impact of the TMOD on the fire sensor is a violation of NRC design control requirements. This Severity Level IV violation is being treated as a Non-Cited Violation consistent with Appendix C of the NRC Enforcement Policy. This violation is entered in the licensees corrective action system as PR 99.9405 (**NCV 50-293/99-04-02**).

c. Conclusions

The inspector identified an EDG TMOD and related safety evaluation did not evaluate the degradation of the fire suppression system. This occurred due to inadequate communications within the engineering department as well as an improper field walk down.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) LER 50-293/99-02: High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems Surveillance Testing Outside Design Bases

This LER documented that the technical specification (TS) upper pressure (1000 psig) for the HPCI and RCIC systems is less than the set point (1126 psig) of the main steam relief vales including tolerances. This upper pressure of 1000 psig was specified in the original TS and remained unchanged since issued in 1972.

The inspector conducted an on-site review of the LER and reviewed the licensee's corrective actions and determined that they were appropriate. The inspector verified that a TS change was submitted and that the surveillance procedures were changed to reflect the upper pressure value to 1126 psig. The HPCI and RCIC systems have been tested to the new pressure limits with satisfactory results. No violations of NRC requirements were identified. This LER is **closed**.

IV. PLANT SUPPORT

S8 Miscellaneous Security Issues (92904)

(Closed) E.I. 98-191-01013, Assessment Area Aids

NRC Inspection Report No. 50-293/98-03 documented multiple examples of equipment failures associated with the licensee's assessment program. As immediate corrective action, all the identified assessment aid deficiencies were corrected. Actions to address the root cause included establishing a CCTV checklist to be performed daily and creating a photo comparison guide. The inspector reviewed the corrective actions taken and planned and determined they were appropriate. A review of assessment aids revealed no problems. This issue is **closed**.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspector met with licensee representatives at the conclusion of the inspection on August 16, 1999. At that time, the purpose and scope of the inspection were reviewed, and the preliminary findings were presented. The licensee acknowledged the preliminary inspection findings.

ATTACHMENT 1

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observation
IP 62706:	Maintenance Rule
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 82301:	Evaluation of Exercises for Power Reactors
IP 92700:	Onsite Follow up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92720:	Corrective Actions
IP 92901:	Follow up - Operations
IP 92902:	Follow up - Maintenance
IP 92903:	Follow up - Engineering
IP 92904:	Follow up - Plant Support
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND UPDATED

Closed

EA 98-277/01014	Failure to establish unavailability measures for the anticipated transient without scram (ATWS) function of the control rod drive system.
EA 98-277/02014	Failure to include in the scope of the maintenance rule the following two structures, systems, or components: Heating, Ventilation, and Air Conditioning (HVAC) for 480V switchgear enclosures in the reactor building, and the firewater system function of providing a backup screenwash supply.
EA 98-277/03014	Failure to complete the periodic evaluation of maintenance rule activities required by 10CFR50.65(a)(3) in a timely manner.
EEI 98-191-01013	Assessment Area Aids
IFI 50-293/98-11-04	AO-220-45 Stroke Time
LER 50-293/99-01	Environmental Enclosures of Motor Control Centers (MCCs) Outside Design Basis
LER 50-293/99-02	High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems Surveillance Testing Outside Design Bases
NCV 50-293/99-04-01	Operator Procedure Use Error
NCV 50-293/99-04-02	Safety evaluation did not evaluate the degradation of the fire suppression system

LIST OF ACRONYMS USED

BECo	Boston Edison Company
CFR	Code of Federal Regulations
CRHEAF	Control Room High Efficiency Air Filtration
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
FRV	Feedwater Regulating Valve
FSAR	Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
IFI	Inspection Follow-Up Item
IR	Inspection Report
LCO	Limiting Condition of Operation
LER	Licensee Event Report
MCC	Motor Control Centers
MO	Motor Operated
MR	Maintenance Request
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NOV	Notice of Violation
NPO	Nuclear Plant Operator
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
PDR	Public Document Room
PNPS	Pilgrim Nuclear Power Station
PR	Problem Report
PWT	Post Work Test
RCA	Radiologically Controlled Areas
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RP	Radiological Protection
RPV	Reactor Pressure Vessel
SBLC	Standby Liquid Control
SBO	Station Blackout
SDC	Shutdown Cooling
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VIO	Violation