UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
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August 2, 2007

Mr. Kevin Bronson
Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360-5508

SUBJECT: PILGRIM NUCLEAR POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000293/2007003

Dear Mr. Bronson:

On June 30, 2007, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Pilgrim Nuclear Power Station. The enclosed integrated inspection report documents the inspection results, which were discussed on July 17, 2007, with you and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents three self-revealing findings of very low safety significance (Green), all of which involved a violation of NRC requirements. However, because of the very low safety significance and because the issues have been entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs), consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Pilgrim.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures and your response (if any) will be available electronically for public inspection in the

NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Raymond J. Powell, Chief
Projects Branch 5
Division of Reactor Projects

Docket No. 50-293
License No. DPR-35

Enclosure: Inspection Report 50-293/07-02
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	05000293/2007003
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	April 1, 2007 through June 30, 2007
Inspectors:	W. Raymond, Senior Resident Inspector (Pilgrim/Seabrook) S. Schneider, Senior Resident Inspector (Millstone/Pilgrim) C. Welch, Resident Inspector T. Burns, Reactor Inspector G. Newman, Reactor Engineer N. Sieller, Project Engineer J. Noggle, Senior Health Physicist R. Rolf, Health Physicist J. Kulp, Reactor Inspector
Approved By:	Raymond J. Powell, Chief Projects Branch 5 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000293/2007003; 04/01/2007-06/30/2007; Pilgrim Nuclear Power Station; Maintenance Effectiveness, Maintenance Risk Assessment and Emergent Work Control, Refueling and Other Outage Activities.

The report covered a 13-week period of inspection by resident inspectors and announced inspections by regional health physics and inservice inspection inspectors. Three Green findings, all of which were non-cited violations (NCVs), were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. A self-revealing non-cited violation of very low safety significance was identified for Entergy's failure to provide adequate work instructions, as required by Pilgrim Technical Specification 5.4.1, "Procedures," to adjust packing on reactor water cleanup valve MO-1201-85, in October 2003. The lack of adequate instructions led to premature packing failure on March 17, 2007, which increased unidentified drywell reactor coolant system leakage, and required a plant shutdown. The direct cause was the failure to apply sufficient compression to the packing when last adjusted in October 2003. Entergy personnel repaired and successfully retested the valve. Entergy entered this issue into their corrective action program and initiated action to develop a packing adjustment procedure, evaluate back seating inaccessible valves, and institute preventive maintenance items to verify the packing gland fastener torque for inaccessible valves.

The finding was more than minor because it adversely affected the equipment performance attribute and objective of the Initiating Events cornerstone of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The finding screened to very low safety significance (Green) per IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," because the maximum observed leak rate did not exceed the Technical Specifications limit for identified reactor coolant system leakage, the finding did not contribute to both the likelihood of a reactor trip and the unavailability of a function of a mitigating system, and the finding did not increase the likelihood of a fire or internal/external flood. This finding has a cross-cutting aspect in the area of Human Performance, Resources, in that Entergy did not ensure that packing adjustment procedures were adequate [H.2(c)]. (Section 1R12)

Green. A self-revealing non-cited violation of very low safety significance was identified for Entergy's failure to properly implement procedure EN-OP-102, "Protective and Caution Tagging," as required by Pilgrim Technical Specification 5.4.1, "Procedures." Specifically, on May 3, 2007, a senior reactor operator approved the removal of a danger tag from 4-HO-50 without ensuring the appropriateness of the component's specified restoration position. As a result, the valve, which was serving as a single point of isolation between the reactor coolant system and the drywell equipment sump, was opened, and approximately six inches of reactor coolant drained from the reactor vessel before the drain path was identified and isolated. Entergy entered this issue into their corrective action program and initiated additional controls and oversight for tagout operations with the potential to interface with the reactor vessel fluid boundary.

The failure to specify the appropriate restoration position constituted a performance deficiency that resulted in an inadvertent decrease of the reactor vessel level totaling six inches. The finding is more than minor because it is associated with the configuration control attribute of the Initiating Events cornerstone, and it affected the associated cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Because this event involved a six inch loss of level, the finding screened to very low safety significance (Green) in accordance with Table 1 of IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The finding had a cross-cutting aspect in the area of Human Performance, Work Control, in that Entergy made a change to a planned work activity, the restoration of 4-HO-50, without fully evaluating the impact of this change on the plant [H.3(b)]. (Section 1R20)

Cornerstone: Mitigating Systems

Green. A self-revealing non-cited violation of very low safety significance was identified for Entergy's failure to implement procedures for testing the analog trip system (ATS) as required by Pilgrim Technical Specification 5.4.1, "Procedures." Specifically, on April 12, 2007, Instrumentation and Controls (I&C) technicians calibrated pressure transmitter PT-263-50A when plant conditions and the requirements of procedure 8.M.2-8.1 did not allow that activity. This resulted in an inadvertent Group 3 primary containment isolation signal which isolated reactor shutdown cooling for 25 minutes. After recovering shutdown cooling, Entergy entered this issue into their corrective action program, conducted a stand down to review this event with I&C personnel, and initiated action to review this and similar procedures which require varying plant conditions.

The finding is more than minor because it is associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance, in accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," because it did not increase the likelihood of a loss of reactor coolant system (RCS) inventory or degrade Entergy's ability to terminate a leak path or add RCS inventory if needed. Throughout this event, adequate thermal margin was maintained since the calculated RCS time-to-boil was greater than 32 hours. This finding has a cross-cutting in the area of Human Performance, Work Practices, in that personnel did not follow the procedure for testing the ATS [H.4(b)]. (Section 1R13)

B. Licensee-Identified Violations

One violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violation and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period at 83 percent power in end-of-cycle coast down. The plant was shutdown on April 6, 2007, to begin refueling outage (RFO) 16. Following completion of RFO 16 activities, the reactor was taken critical on May 8, 2007, and the plant returned to 100 percent power on May 12, 2007. The plant operated at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

Partial System Walkdowns

a. Inspection Scope (4 samples)

The inspectors completed a partial system review of the risk significant systems listed below to determine whether the systems were correctly aligned to perform their designated safety functions. The reviews occurred during periods when a redundant train or system was out-of-service for maintenance and/or testing, or following restoration of the system or train from maintenance. The position of key valves, breakers, and control switches required for system operability were verified by field walkdown and/or review of the main control board indications. To ascertain the required system configuration, the inspectors reviewed plant procedures, system drawings, the Updated Final Safety Analysis Report (UFSAR), and Technical Specification (TS). The references used for this review are listed in the attachment to this report.

- Residual Heat Removal (RHR) system shutdown cooling alignments;
- RHR system augmented fuel pool cooling Mode 1 lineup;
- Pre-startup alignment checks for Core Spray trains A & B; and
- "B" Emergency Diesel Generator (EDG) system alignment during maintenance on "A" EDG.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Fire Protection Inspection

a. Inspection Scope (11 samples)

The inspectors toured selected areas of the plant to observe conditions related to: (1) transient combustibles and ignition sources; (2) fire detection systems; (3) manual firefighting equipment and capability; and (4) passive fire protection features. The inspectors verified adequate material condition of active and passive fire protection systems features and their operational lineup and readiness. The inspectors also reviewed the applicable fire hazard analysis fire zone data sheets. The inspectors verified that the licensee addressed fire protection deficiencies in the corrective action program. The references used for this review are listed in the Attachment to this report. The areas of the plant selected were:

- Fire Zone 1.20, Refueling Floor;
- Fire Zone 1.30, Drywell;
- Fire Zone 1.30A, Torus Compartment;
- Fire Zone 1.32, Main Steam Tunnel;
- Fire Zone 2.8, Condensate Pump Area;
- Fire Zone 2.8A, Condensate Demineralizer Areas;
- Fire Zone 2.8B, Condenser Vacuum Pump Room;
- Fire Zone 2.9, “A” Train Feedwater Heater Bay;
- Fire Zone 2.9A, “A” Train Condenser Bay;
- Fire Zone 2.10, ”B” Train Feedwater Heater Bay; and
- Fire Zone 2.10A, “B” Train Condenser Bay.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope (1 Internal Flood Protection sample)

The inspectors reviewed protective measures in place to protect against internal flooding of the “A” and “B” reactor building closed cooling water (RBCCW) rooms during periods of maintenance on the “A” RBCCW heat exchanger. The inspectors performed visual inspections of the water tight door separating the “A” and “B” compartments, curbing around switchgear, and the de-watering lines from each compartment to the torus room. Flood barriers and level switches were inspected to determine whether they could perform their intended functions.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope (1 sample)

The inspectors reviewed the licensee’s heat exchanger testing, data evaluation, and performance trending for the “B” RHR heat exchanger to determine if the licensee was effectively monitoring the heat removal capacity and capability to fulfill the required safety function. The review included test results from procedure 8.5.3.14.2, “RHR Heat Exchanger Thermal Performance Test,” and calculation M710, “Heat Exchanger Performance Testing,” completed during RFO 16.

The inspectors assessed whether test results were compared against established acceptance criteria; if differences between plant conditions and design conditions were accounted for; the adequacy of the frequency of testing and inspections; and if test results met the acceptance criteria and demonstrated the required salt service water flow could be achieved without exceeding the design basis pressure drop across the heat exchangers. The inspectors also walked down the “B” RHR heat exchanger to assess material conditions and verified that discrepancies were evaluated and entered into the corrective action program.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope (7 samples)

The purpose of this inspection is to assess the effectiveness of the licensee’s Inservice Inspection (ISI) program for monitoring degradation of the reactor coolant system (RCS) boundary, risk significant piping system boundaries, and the containment boundary. In addition, the inspectors reviewed the results of dissimilar metal weld examination activities specific to the welding of safe ends to reactor pressure vessel (RPV) nozzles. The inspectors assessed the ISI activities using requirements and acceptance criteria specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, and applicable NRC Regulatory Requirements.

The inspectors selected a sample of non-destructive examination (NDE) activities for observation and documentation review for compliance with the requirements of ASME Section XI. The sample selection was based on the inspection procedure objectives, sample availability, and risk priority of those components and systems where degradation could result in a significant increase in risk of core damage. The inspectors verified by documentation review that test examiner’s qualifications were current and in accordance with the ASME Code requirements. Also, the inspectors reviewed examiner qualifications for use of the performance demonstration initiative (PDI) ultrasonic test procedures to examine the recirculation inlet nozzle to safe end welds. The inspectors also evaluated the licensee’s effectiveness in resolving relevant indications identified during the observed ISI activities. The inspectors’ observation and documentation review of non-destructive testing included the following four samples:

- Ultrasonic testing (automatic PDI Qualified) of RPV nozzle to safe end welds 2R-N2A-1, 2R-N2B-1, 2R-N2C-1, 2R-N2K-1 and (manual PDI Qualified) N9B-1;
- Ultrasonic testing (manual) of butt welds 14-B-18 and 14-B-19, pipe to pipe and pipe to fitting in the core spray system;
- Magnetic particle test of integral attachment welds to HE-26-175HL1 (1), high pressure core injection (HPCI) system; and
- Liquid penetrant test of integral attachments to the reactor recirculation system piping 2R-N1B-14HL2(4).

The inspectors selected the remote visual examination (VT-1 and VT-3) of the steam dryer for review of the in-vessel visual inspection (IVVI) activity to evaluate the effectiveness of the vessel internals inspection program. The inspectors reviewed portions of the remote IVVI of the reactor steam dryer base metal and structural welds to evaluate examiner skill, test equipment performance, examination technique, and inspection environment (water clarity) to verify the licensee's ability to identify and characterize observed indications. This review constituted one inspection sample.

The inspectors selected two ASME Section XI repair/replacement plans for review where welding on a pressure boundary had been completed. The review was performed to evaluate control of the welding process and that welding examinations were performed in accordance with the ASME code requirements. As a result of the inspector's review, three condition reports (CRs) (CR-PNP-2007-02033, 02035 and 02051) were initiated to identify and document the inspector's observations. The two ASME Section XI repair/replacement activities reviewed were:

- Maintenance Request (MR) 07105779 R1, repair leaking valve disc seat by weld build up, RHR pump "C" torus suction valve, MO-1001-7C; and
- MR 07106245 R0, replace valve 1-HO-64, valve will not hold as boundary valve, main steam line, system #1, "D" low point drain.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

Resident Inspector Quarterly Review

a. Inspection Scope (1 sample)

On May 7, 2007, the inspectors observed just-in-time (JIT) training of an operating crew preparing for the impending reactor / reactor plant startup and main turbine overspeed testing. Training was comprised of both classroom and simulator instruction. The JIT training covered the Power Maneuver Plan, approach to critical, reactor criticality, reactor low power operations, placing the main turbine on-line and main turbine overspeed testing. Industry and plant specific operating experience, related to the covered activities, was also discussed.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope (2 samples)

The inspectors reviewed the follow-up actions for selected system, structure, or component (SSC) issues and reviewed the performance history of these SSCs to assess the effectiveness of Entergy's maintenance activities. The inspectors reviewed Entergy's corrective actions for these issues in accordance with Entergy procedures and the requirements of 10 CFR 50.65(a)(1) and (a)(2), "Requirements for Monitoring the

Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, system health reports, and corrective actions that were taken or planned to verify whether the actions were reasonable and appropriate. The inspectors attended licensee meetings and reviewed licensee plans to address the systems in maintenance rule a(1) status. The following issues were reviewed:

- Reactor Building Closed Cooling Water pump P202D low total dynamic head, CR 20070719; and
- Reactor Water Cleanup (RWCU) valve packing failure on MO-1201-85, CR 20070949.

b. Findings

Introduction: A Green self-revealing non-cited violation (NCV) of very low safety significance was identified for Entergy's failure to provide adequate work instructions, as required by TS 5.4.1, "Procedures," to adjust packing on RWCU valve MO-1201-85, in October 2003. The lack of adequate instructions led to premature packing failure on March 17, 2007, which increased unidentified drywell RCS leakage and required a plant shutdown. The direct cause was the failure to apply sufficient compression to the packing when last adjusted in October 2003.

Description: At approximately midnight on March 17, 2007, operations personnel noted an elevated drywell floor sump leakage rate (RCS unidentified leakage). The calculated leak rate had risen from 0.58 to 0.65 gallons per minute (gpm). Since the leak rate continued to increase, Entergy made the decision to shutdown and locate the source of leakage. Shutdown commenced at 12:30 p.m. and the reactor was fully shutdown at 4:55 p.m. Drywell unidentified leak rate increased to a maximum value of 2.59 gpm, and then decreased as the reactor was cooled down and depressurized. Drywell leakage remained below the TS 3.6.C.1 limits for total (25 gpm) and unidentified (5 gpm) leak rate. Entergy personnel entered the drywell on March 18, 2007, and identified that the source of leakage was a packing failure on RWCU valve MO-1201-85. Entergy personnel repaired the valve and successfully retested it.

During the subsequent root cause evaluation, Entergy identified that the packing gland bolts were hand, not wrench, tight. The valve, last repaired in 1984, had minor packing adjustments in 1984, 1987, and 2003. The packing adjustments were performed as "skill of the craft." A review of the 2003 work plan (MR 03117384) identified that no torque requirements were provided for the packing gland nuts. A contributing cause identified by Entergy was the absence of a valve packing program or preventive maintenance (PM) item to periodically verify the packing gland nut torque for valves in the drywell, which are inaccessible during plant operations. Entergy's root cause identified several opportunities, based on operating experience and plant assessment recommendations, to institute a robust valve packing program at Pilgrim. However, in each case Entergy deferred establishing a formal program based on prior success and higher competing priorities.

Analysis: The performance deficiency associated with this finding is that Entergy did not provide adequate work instructions for the packing adjustment of MO-1201-85 which led to premature failure of the valve's packing. As a result, drywell unidentified RCS leakage increased to a point which necessitated a plant transient (shutdown) to identify and correct. The finding was more than minor because it adversely affected the equipment performance attribute and objective of the Initiating Event cornerstone of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The finding screened to very low safety significance (Green) per Inspection Manual Chapter (IMC) 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," because the maximum observed leak rate did not exceed TS limit for identified RCS leakage, the finding did not contribute to both the likelihood of a reactor trip and the unavailability of a function of a mitigating system, and the finding did not increase the likelihood of a fire or internal/external flood.

The performance deficiency has a cross-cutting aspect in the area of Human Performance, Resources, in that Entergy did not ensure that packing adjustment procedures were adequate.

Enforcement: Technical Specification 5.4.1, "Procedures," states, in part, that written procedures shall be established, implemented and maintained as recommended in Regulatory Guide (RG) 1.33, which includes procedures for valve maintenance. Contrary to the above, in October 2003, Entergy did not provide the mechanics adequate instructions to adjust the packing on the MO-1201-85 valve. Entergy procedure 3.M.4-10 (Revision 33), "Valve Maintenance," was included in the October 2003 work package (MR 03117384). However, the procedure was only used to capture post work test requirements and not to provide direction for the packing adjustment which was left to the skill of the craft. The lack of instruction resulted in failure to adequately compress the packing and establish the required torque on the packing gland nuts which ultimately resulted in failure of the packing on March 17, 2007. Entergy initiated action to develop a packing adjustment procedure, evaluate back seating inaccessible valves, and institute preventive maintenance items to verify the packing gland fastener torque for inaccessible valves. Because the finding was of very low safety significance (Green) and has been entered into Entergy's corrective action program (CR 20070949), this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000293/2007003-01, Failure to provide adequate instructions for adjusting MO-1201-85 packing resulted in premature packing failure.)**

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope (6 samples)

The inspectors evaluated Entergy's risk management for planned and emergent work for both on-line and shutdown plant conditions. The inspectors reviewed maintenance risk evaluations, defense-in-depth work sheets, work schedules, risk management related corrective actions, and control room logs to verify that concurrent planned and emergent maintenance and surveillance activities did not adversely affect plant risk and/or the defense-in-depth critical safety function strategies. The inspectors evaluated whether Entergy took the necessary steps to control work activities, minimize the probability of initiating events, and maintain the functional capability of mitigating systems. The inspectors assessed Entergy's risk management actions during plant walkdowns and discussed risk management with maintenance, engineering, and operations personnel, as applicable, for the reviewed activities. References used for the inspection are identified in the Attachment to this report.

- Elevated outage risk condition on April 11, 2007, due to reduced electrical power availability in support of maintenance and testing of the station blackout diesel generator, bus A5 and bus A8;
- Elevated outage risk condition on April 12, 2007, following the inadvertent isolation of shutdown cooling, CR 200701503, during performance of surveillance procedure 8.M.2-8.1;
- Elevated outage risk condition on April 18, 2007, due to heavy load lifts within the drywell for removal of the "B" reactor recirculation pump and motor;
- Elevated outage risk condition on April 19, 2007, due to isolation of the shutdown cooling lineup in support of maintenance and testing of residual heat removal isolation valves MO-1001-47 and 50;
- Elevated outage risk condition on April 24-25, 2007, due to isolation of shutdown cooling during loss of power testing per 8.M.3-1; and
- Elevated on-line risk condition May 14-19, 2007, due to emergent work and testing of the "A" EDG.

b. Findings

Introduction: A Green self-revealing NCV was identified for Entergy personnel's failure to follow a procedure for testing the analog trip system (ATS) as required by TS 5.4.1, "Procedures." Specifically, on April 12, 2007, Instrumentation and Control (I&C) technicians calibrated pressure transmitter (PT) PT-263-50A when plant conditions, and the requirements of procedure 8.M.2-8.1, "Calibration of ATS Transmitters Rack C2205," did not allow that activity. This resulted in a primary containment isolation system (PCIS) Group 3 signal which isolated reactor shutdown cooling for 25 minutes.

Description: On April 12, 2007, with reactor cooling provided by the "B" RHR system operating in shutdown cooling, I&C technicians began a calibration of pressure transmitter PT-263-50A using procedure 8.M.2-8.1. The technicians' work assignment had been to perform procedure 8.M.1-30, "ATWS System Calibration Test," but the technicians started 8.M.2-8.1 in error. Procedure 8.M.2-8.1, Step 7.0, and Attachment 1 Step [6], required that the RHR system not be in shutdown cooling for calibration of PT-263-50A. The technicians did not adequately verify this requirement was met before proceeding with the calibration.

When the technicians began the calibration of PT-263-50A, a PCIS Group 3 signal was generated at 9:29 p.m. The PCIS Group 3 signal isolated the single train suction path for shutdown cooling and tripped the operating "B" RHR pump. The operators responded by entering procedure 2.4.25, "Loss of Shutdown Cooling." After determining the cause of the event, the operators reset the Group 3 isolation and restored Shutdown cooling at 9:53 p.m. Operators were able to reestablish a suction path and restart the "B" RHR pump in approximately 25 minutes. Throughout this event, adequate thermal margin was maintained since the calculated RCS time-to-boil was greater than 32 hours. The actual reactor coolant temperature rise was less than two degrees Fahrenheit.

Entergy entered this event into their corrective action program as CR 200701503. In the associated apparent cause analysis, Entergy determined the cause of the event was the failure to implement procedure 8.M.2-8.1. Specifically, the technicians did not fully review the procedure and precautions prior to calibrating PT-263-50A.

Analysis: The performance deficiency associated with this finding is that Entergy did not adequately implement a procedure for the calibration of the ATS. Calibrating PT-263-50A during reactor conditions, contrary to procedure 8.M.2-8.1, resulted in a loss of shutdown cooling for approximately 25 minutes. The finding is more than minor since it is associated with the Mitigating Systems cornerstone attribute of human performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors conducted a SDP Phase 1 screening of the finding in accordance with IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process.” The finding was determined to be of very low safety significance (Green) because, although the finding resulted in there being less than one loop of RHR in shutdown cooling operation, it did not increase the likelihood of a loss of RCS inventory, degrade Entergy’s ability to terminate a leak path or add RCS inventory if needed, or degrade the ability to recover decay heat removal.

This finding is related to the cross-cutting area of Human Performance, Work Practices, in that Entergy did not adequately implement a procedure for the calibration of the ATS.

Enforcement: Technical Specification 5.4.1, “Procedures,” Section A, requires that written procedures be established, implemented and maintained as recommended in RG 1.33, which includes procedures for tests and calibrations. Procedure 8.M.2-8.1, “Calibration of ATS Transmitters Rack C2205,” is a procedure used by Entergy for the calibration of ATS instruments, including PT-263-50A. Procedure 8.M.2-8.1, Step 7.0, and Attachment 1, Step [6], required that the RHR system not be in shutdown cooling during calibration of PT-263-50A. Contrary to the above, on April 12, 2007, Entergy proceeded with the calibration of PT-263-50A despite the RHR system being in shutdown cooling. When Entergy began the calibration of PT-263-50A, a PCIS Group 3 signal was generated which isolated the single train suction path for RHR, tripped the operating “B” RHR pump and resulted in the loss of shutdown cooling for approximately 25 minutes. After recovering shutdown cooling, Entergy conducted a stand down to review this event with I&C personnel and initiated action to review this and similar procedures which require varying plant conditions. Because the finding was of very low safety significance (Green) and has been entered into Entergy’s corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000293/2007003-02, Failure to follow procedures resulted in a loss of shutdown cooling**.)

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (4 samples)

The inspectors reviewed selected operability determinations to assess the adequacy of the evaluations, the use and control of compensatory measures, compliance with the TS, and the risk significance of the issues. The inspectors used the TS, UFSAR, associated design basis documents, and the additional references listed in the Attachment to this report. The inspectors reviewed:

- CR 200701172, Secondary Containment Ventilation dampers AO-78/79/80 did not fully close during secondary containment testing;
- CR 200701277, B residual heat removal heat exchanger flange leak at 3 gallons per minute in shutdown cooling;
- CR 200701229, Significant kW oscillations on “A” Emergency Diesel Generator during performance of 8.9.1 monthly EDG testing; and
- CR 200701880, quantity of sludge removal from torus exceeded design basis loading of the emergency core cooling system (ECCS) suction strainers.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (6 samples)

The inspectors reviewed post-maintenance test (PMT) activities on risk significant systems to determine whether the effect of the test on the plant had been evaluated adequately, the test was performed in accordance with procedures, the test data met the required acceptance criteria, and the test activity was adequate to verify system operability and functional capability following maintenance. The inspectors confirmed that systems were properly restored following testing and that discrepancies were appropriately documented in the corrective action process. References used during this review are listed in the attachment to this report.

- MSIV 2A (AO-203-2A) repair/rebuild, MRs 19800194 and 07107233;
- MSIV 2C (AO-203-2C) repair/rebuild, MR 02120627;
- MO-1001-7C valve repair, MR 07105779;
- Reactor Core Isolation Cooling (RCIC) Overhaul, MRs P9400991, 06103659 and 06105406;
- E-207B, “B” residual heat removal heat exchanger flange leak repair, MR 07105865; and
- P202D, RBCCW pump “D” rebuild, MR 07102534.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope (1 sample)

Periodic review of RFO 16 Work Plan and Outage Risk

The inspectors, on a routine basis, reviewed the refueling outage work plan and daily shutdown risk assessments to verify Entergy addressed the outage impact on defense-in-depth for the five shutdown critical safety functions: electrical power availability, inventory control, decay heat removal, reactivity control, and containment. Periodic risk updates, accounting for schedule changes and unplanned activities, were also reviewed. The inspectors’ review focused on verifying the licensee had provided adequate defense-in-depth for each safety function, and/or implemented planned contingencies to minimize the overall risk where redundancy was limited or not available. Detailed risk reviews for specific high risk periods and activities are documented in section 1R13 of this report.

Monitoring of Shutdown Activities

The inspectors observed operators perform portions of the reactor shutdown, initial noble metals chemistry application, and plant cooldown to assess operator performance with respect to communications, command and control, procedure adherence, and compliance with TS cooldown limits. Upon shutdown, the inspectors also conducted inspection walkdowns of plant areas not normally accessible during plant power operations (drywell, condenser bay, torus, and main steam tunnel) to verify the integrity of structures, piping and supports, and to confirm systems appeared functional.

Clearance Activities

The inspectors reviewed a sample of risk significant clearance activities and verified tags were properly hung and/or removed, equipment was appropriately configured per the clearance requirement, and that the clearance did not impact equipment credited to meet the shutdown critical safety functions. The following clearances were reviewed:

- ECCS clearances: 1-RFO-16-03643, 1-RFO-16-04027, 1-RFO-16-04023, 1-RFO-16-04024; and
- Recirculation system clearances: 1-RFO-16-00142 through 1-RFO-16-00153.

RCS Instrumentation

The inspectors periodically observed and verified by diverse means that associated instruments for the reactor/refueling cavity/spent fuel pool (SFP) water level, the reactor coolant and spent fuel pool temperature, and the operating residual heat removal system were functioning properly and accurately.

Electrical Power

The inspectors verified that the status of electrical systems met TS requirements and the licensee's outage risk control plan. The inspectors verified that compensatory measures were implemented when electrical power supplies were impacted by outage work activities. The inspectors verified that credited backup power supplies were available.

RHR and SFP System Monitoring

The inspectors observed the SFP and reactor RHR system status and operating parameters to verify that the cooling systems operated properly. Verification included periodic review of the SFP and reactor cavity level, temperature, and RHR system flow. Partial system walkdowns, to verify proper system configuration, were periodically performed for both the normal RHR alignment and augmented SFP cooling lineup.

Inventory Control

The inspectors reviewed Entergy's actions to establish, monitor, and maintain the proper water inventory in the reactor vessel and spent fuel pool. The inspectors reviewed the plant system flow paths and configurations established for reactor makeup and verified the configurations were consistent with the outage plan. The inspectors reviewed Entergy's response and root cause determination for the inadvertent reactor vessel level decrease that occurred on May 3, 2007.

Foreign Material Exclusion (FME)

The inspectors reviewed implementation of licensee procedures for FME control for the open reactor vessel, reactor cavity, and SFP. Entergy's actions to identify, document, and resolve FME events/issues were reviewed by the inspectors.

Control of Heavy Loads

The inspectors reviewed licensee actions to control the lift of heavy loads during the outage. The review included activities related to the heavy loads associated with the disassembly of the drywell and the reactor vessel, the installation of the in-vessel inspection platform, and the replacement of the "B" reactor recirculation pump motor and pump. The inspectors reviewed licensee actions to manage the increased risk during these activities and to implement compensatory measures to protect the integrity of systems important to safe shutdown and cooling of the reactor and SFP. This review included consideration of industry operating experience and licensee commitments to NRC regulatory guidance.

Containment Control

The inspectors reviewed licensee activities during the outage to control primary and secondary containment and to clean and prepare the containment for closure prior to plant restart. The inspectors performed periodic tours of the drywell and torus to review the control of work activities and containment conditions. The inspectors conducted walkdowns of the drywell prior to reactor startup to review licensee cleanup and demobilization controls in areas where work was completed to assure that tools, materials and debris were removed. This review focused on the removal of debris which might impact the performance of the safety systems.

The inspectors observed and/or reviewed data for a sample of primary containment penetration local leak rate tests (LLRT). The inspectors also reviewed post-outage overall containment leakage for compliance with TS 4.7.A.2, "Primary Containment Integrity."

Fuel Shuffle Activities and Reactivity Control

The inspectors verified that refueling activities were conducted in accordance with Entergy procedure 4.3, "Fuel Handling;" and verified, on a sampling basis, that requirements for core alteration were met. The inspectors observed licensee actions and activities from the control room and the refueling floor at various times during core alterations to assure core reactivity was controlled. The inspectors verified that fuel movement was accomplished and tracked in accordance with the fuel movement schedule. The inspectors also verified licensee action to meet the requirements of TS 3.10 for core alterations, including the requirements for core monitoring using the source range monitors and the functional checks of the refueling interlocks. The inspectors observed communications and the coordination of activities between the control room and the refueling floor while fuel handling activities were in progress. The inspectors observed performance of Entergy procedures 4.5, "Reactor Core Fuel Verification," and 9.16.2, "Subcritical Demonstration." The inspectors reviewed the core mapping video recording and verified the as-loaded core configuration matched the designed core reload configuration.

Monitoring Heatup and Startup Activities

The inspectors observed and/or reviewed heatup and startup activities during the period of May 5 through May 10, 2007. The inspection consisted of control room observations, plant walkdowns, and a review of control board indicators, operator logs, plant computer information, and station procedures: 2.1.8.5, "Reactor Vessel Pressurization and Temperature Control For Class 1 System Leakage Test;" 2.1.1, "Startup from Shutdown;" and 2.1.14, "Station Power Changes." The inspectors observed operator actions including the preparations for the approach to critical, reactor critical operations, the in-sequence shutdown margin determination accomplished in accordance with procedure 9.16.1, "In-Sequence Critical for Shutdown Margin Demonstration," low power operations, and the synchronization of the main turbine generator to the electrical grid. The inspectors observed plant restart and power ascension operations in accordance with procedures 2.1.1 and 2.1.14 to verify that TS, license conditions, and other requirements for mode changes were met.

Problem Identification and Resolution

The inspectors verified that Entergy was identifying outage related issues and had entered them into the corrective action program. The inspectors reviewed a sample of the corrective actions to verify they were appropriate to resolve the issues. The references used in this review are listed in the attachment to this report.

b. Findings

Introduction: A Green self-revealing NCV of TS 5.4.1, "Procedures," was identified for the failure to adequately implement the Entergy protective tagging process in accordance with procedure EN-OP-102, "Protective and Caution Tagging." Specifically, on May 3, 2007, a senior reactor operator (SRO) approved a tag removal for valve 4-HO-50 without ensuring the component's restoration position was appropriately specified. This resulted in a drain path being established from the RCS to the drywell equipment sump. Approximately six inches of reactor coolant was lost from the reactor vessel before the drain path was identified and isolated.

Description: On May 3, 2007, Pilgrim was shut down with reactor water level near the vessel flange and the core fully loaded. Main steam line operating vent valve 4-HO-50 was danger tagged in the closed position and serving as a single point of isolation between the RCS and the drywell equipment sump. In preparation for a reactor vessel pressurization test, operators were hanging test and maintenance tags on all valves affected by the test, including valve 4-HO-50. Since 4-HO-50 had previously been danger tagged, operators were to remove the danger tag, in accordance with approved restoration instructions, and hang a test and maintenance tag in its place.

Restoration instructions were generated by computer, using a program that selects each component's restoration position based on its normal operating position. Entergy procedure EN-OP-102 allows for restoration instructions to be annotated when the desired valve position is different from the normal operating position originally specified.

EN-OP-102 further states that it is the responsibility of the operations supervisor approving the tag removal to ensure appropriate restoration instructions have been annotated. Because the normal operating position of 4-HO-50 is open, but the reactor vessel pressurization test required it to be closed, the shift tagging coordinator made a pen and ink change to the instructions to indicate the valve should be left closed.

During the pre-job brief, the SRO responsible for executing the test valve lineup questioned the pen and ink change the tagging coordinator had made, and erroneously determined the valve should be placed in the open position. The SRO communicated this change back to the tagging coordinator and annotated the restoration instructions to indicate the valve should be placed in the open position. Neither the SRO nor the tagging coordinator recognized that, given the current valve lineup and reactor vessel water level, 4-HO-50 was serving as a single point of isolation between the RCS and the drywell equipment sump, and that opening the valve would initiate flow from the vessel to sump. When the operators opened the valve in accordance with the restoration instructions, the reactor vessel began to drain at a rate of approximately four gallons per minute. The drain down continued for about two hours, until reactor operators recognized the lowering reactor vessel level, identified the drain path, and closed 4-HO-50. A total of six inches of reactor vessel level was lost during the two hour event.

<u>Analysis</u>: The performance deficiency associated with this finding is that Entergy did not ensure that an appropriate restoration position was specified for valve 4-HO-50. The failure to specify an appropriate restoration position resulted in an inadvertent reactor vessel level decrease. The finding is more than minor because it is associated with the configuration control attribute of the Initiating Events cornerstone, and it affected the associated cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. The inspectors conducted a Phase 1 screening of the finding in accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." Table 1 of IMC 0609, Appendix G specifies that an inadvertent loss of level in a boiling water reactor is considered a loss of control, and requires a Phase 2 or Phase 3 assessment, if more than two feet of RCS inventory were lost. Since this event involved only a six inch loss of level, a quantitative assessment was not required, and the finding screened as having very low safety significance (Green).

The performance deficiency has a cross-cutting aspect in the area of Human Performance, Work Control, in that Entergy made a change to a planned work activity, the restoration of 4-HO-50, without fully evaluating the impact of this change on the plant.

<u>Enforcement</u>: Technical Specification 5.4.1 requires that procedures be established, implemented, and maintained covering the applicable procedures in RG 1.33, Appendix A. Paragraph 1.c of Appendix A requires procedures for equipment control, including tagging. Entergy's tagging procedure, EN-OP-102, "Protective and Caution Tagging," states that an operations supervisor is responsible for ensuring that appropriate restoration positions have been annotated for tagout removals. Contrary to this requirement, on May 3, 2007, an SRO approved an inappropriate restoration position for 4-HO-50, which resulted in an inadvertent reactor vessel level decrease. Entergy initiated additional controls and oversight for tagout operations with the potential to interface with the reactor vessel fluid boundary. Because this finding is of very low

safety significance and has been entered into the licensee's corrective action program (CR-PNP-2007-02326) this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000293/2007003-03, Inadvertent Decrease in Reactor Vessel Level Due to Personnel Error)**

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (10 Samples)

The inspectors observed surveillance tests and/or reviewed test results to determine whether the test acceptance criteria was consistent with TS, that the test was performed in accordance with the written procedure, the test data was complete and met procedural requirements, and the components were capable of performing their intended safety functions. Additional references used for this review are listed in the attachment to this report. The inspectors observed/reviewed:

- 8.M.3-1, "Special Test for Automatic ECCS Load Sequence of Diesels and Shutdown Transformer with Simulated Loss of Offsite Power and Special Shutdown Transformer Load Test;" Revision 51;
- 8.7.3, Secondary Containment Leak Rate Test, Revision 55;
- 8.M.2-1.5.4, RHR Isolation Valve Control Test A Inboard, Revision 22;
- 8.M.2-1.5.5, RHR Isolation Valve Control Test B Outboard, Revision 22;
- 8.5.4.4, HPCI Valve (Quarterly) Operability Test, Revision 47;
- 8.5.4.9, High Pressure Coolant Injection Turbine Overspeed Trip Test, Revision 20
- 8.5.4.3, High Pressure Coolant Injection Operability Demonstration and Flow Rate Test at 150 PSIG, Revision 48;
- 8.5.4.1, High Pressure Coolant Injection (HPCI) System Pump and Valve Quarterly and Biennial Comprehensive Operability, Revision 102;
- 8.7.1.6, "Local Leak Rate Testing of the Main Steam Isolation Valves, Revision 24; and
- 8.5.6.2, Special Test for ADS System Manual Opening of Relief Valves, Revision 36.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope (1 sample)

The inspectors reviewed temporary modification 06-1-056, Noble Metals Injection (TP06-028) to verify that the licensing bases and performance capability of the associated risk significant system had not been degraded through the modification. A walkdown was performed to determine whether equipment was installed in accordance with instructions. The inspectors reviewed applicable drawings and procedures to determine whether they reflected the temporary modifications. The references used for this review are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (9 samples)

During April 23-28, 2007, the inspectors conducted the following activities to verify that the licensee was implementing physical, engineering, and administrative controls for access to high radiation areas, and other radiologically controlled areas, and that workers were adhering to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10 CFR 20, TS, and the Entergy procedures.

(1) Radiation work permits (RWPs) that provide access to exposure significant areas of the plant including high radiation areas were reviewed. Specified electronic personal dosimeter alarm set points were reviewed with respect to current radiological condition applicability and workers were queried to verify their understanding of plant procedures governing alarm response and knowledge of radiological conditions in their work area.

(2) There were no RWPs for airborne radioactivity areas with the potential for individual worker internal exposures of >50 mrem Committed Effective Dose Equivalent (CEDE). However, the potential for internal exposures due to transuranic radionuclides in the occupational workplace was reviewed in detail. Prior to the refueling outage in February 2007, the Chemistry Department identified traces of transuranics in reactor water and later air samples from the drywell continuous air monitor, C-19, which sporadically indicated traces of transuranic radioactivity. Based on this information, a CR was written and the licensee entered procedure EN-142, "Failed Fuel Response," which required actions from various plant departments including Radiation Protection. Based on the possibility of transuranics in open systems associated with reactor water and the potential internal exposure hazard, the Radiation Protection Manager instituted a Standing Order before the beginning

of the refueling outage requiring all air samples taken inside primary containment to be measured for alpha radiation and analyzed for the most restrictive transuranic radionuclide annual limit of intake.

The inspectors took independent contamination surveys inside primary containment and verified the presence of alpha contaminants. The inspectors also reviewed the air sample record for the primary containment and verified that these air samples were counted for alpha contaminants, that the counting statistics were adequate to measure fractions of a transuranic airborne radioactivity area, and that there was only one low-level and short lived transuranic airborne radioactivity area identified during the refueling outage.

Due to the continuing possibility of transuranics in plant systems, the Radiation Protection Manager initiated another CR (CR-PNP-2007-2150) to assess the radiation protection program's ability to survey and protect occupational workers during future work activities associated with applicable open plant systems.

(3) During the refueling outage, internal dose assessments were reviewed. The assessments did not indicate any internal exposure in excess of 50 mrem CEDE.

(4) During April 23-28, 2007, the following radiologically significant work activities were selected; the radiological work activity job requirements were reviewed; and work activity job performance was reviewed with respect to the radiological work requirements.

- Refueling activities;
- "B" recirculation motor replacement;
- NDE of various reactor vessel nozzles;
- Under vessel preparations for low power range monitor replacement;
- Installation of temporary shielding in the drywell; and
- Torus diving inspection activities.

(5) During observation of the work activities listed in (4) above, the adequacy of surveys, job coverage and contamination controls were reviewed.

(6) The torus diving activities involved significant dose gradients and the licensee's actions in monitoring the whole body and extremities during this work, was reviewed.

(7) The inspectors verified the adequacy of locking entrances to accessible locked high radiation areas in the plant and reviewed the inventory and control of keys to these locks under the control of Radiation Protection and Operations.

(8) During observation of the work activities listed in (4) above, radiation worker performance was evaluated with respect to the specific radiation protection work requirements and their knowledge of the radiological conditions in their work areas.

(9) During observation of the work activities listed in (4) above, radiation protection technician work performance was evaluated with respect to their knowledge of the radiological conditions, the specific radiation protection work requirements and radiation protection procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (4 samples)

During April 23-28, 2007, the inspectors conducted the following activities to verify that the licensee was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and the licensee's procedures.

(1) The following highest exposure work activities for RFO 16 were selected for review:

- Refueling activities;
- "B" recirculation motor replacement;
- NDE of various reactor vessel nozzles;
- Under vessel preparations for low power range monitor replacement;
- Installation of temporary shielding in the drywell; and
- Torus diving inspection activities.

(2) With respect to the work activities listed in (1) above, these job sites were observed to evaluate if surveys and ALARA controls were implemented as planned.

(3) With respect to the work activities listed in (1) above, radiation worker and radiation protection technician performance was observed during the performance of these work activities to demonstrate the ALARA principles.

(4) During the past year, dosimetry records were screened for declared pregnant workers, with one identified. The exposure results and monitoring controls were examined with respect to requirements in 10 CFR 20.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope (3 samples)

The Initiating Events cornerstone performance indicator (PI) data for unplanned scrams per 7,000 critical hours; unplanned scrams with loss of normal heat removal; and unplanned power changes per 7,000 critical hours were reviewed to assess the completeness and accuracy of the reported information. Specifically, PI data for the year 2006 and first quarter 2007 was reviewed and compared to information contained in NRC inspection reports, Licensee Event Reports (LERs), and operator logs.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Daily Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” the inspectors performed a screening of each item entered into the licensee’s corrective action program. This review was accomplished by reviewing printouts of each condition report, attending daily screening meetings, and/or accessing the licensee’s database. The purpose of this review was to identify conditions such as repetitive equipment failures or human performance issues that might warrant additional follow-up.

b. Findings and Observations

No findings of significance were identified.

.2 Corrective Action Program Semi-Annual Trend Review

a. Inspection Scope (1 sample)

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspectors performed the semi-annual trend review to identify trends, either Entergy or NRC identified, that might indicate the existence of a more significant safety issue. Included within the scope of this review were CRs from October 2006 through May 2007, the 2007 first quarter system health report, and the Pilgrim Station Quarterly Trend Report for the fourth quarter of 2006. The Pilgrim Station Quarterly Trend Reports for the first and second quarter of 2007 had not been issued at the time of the inspection and were not included in the review.

b. Findings and Observations

No findings of significance were identified. Although several trends were identified, none of these trends indicated the presence of broader safety issue. The trends identified by the inspectors were recognized by Entergy. Most of the trends have been evaluated using apparent cause investigations or are subject to increased monitoring. The inspectors identified that the adverse trend noted by Entergy in the area of contamination control has not improved. Adverse trends identified but not captured in the Pilgrim Station Quarterly Trend Report (Fourth Quarter 2006) were noted in the areas of FME controls and the increased concentration of iron in the reactor feedwater system. The adverse trend in feedwater iron is captured in the First Quarter Adverse Trend Condition Report List.

.3 Occupational Radiation Safety Cornerstone

a. Inspection Scope

The inspectors reviewed four CRs associated with the radiation protection program and the refueling outage which were initiated between February and April 2007. The inspectors verified that problems identified by these CRs were properly characterized and that applicable causes and corrective actions were identified, commensurate with the safety significance of the radiological occurrences.

b. Findings and Observations

No significant findings or observations were identified.

.4 Inservice Inspection

a. Inspection Scope (2 samples for 1R08)

CRs were initiated for a linear indication identified in the steam dryer divider plate anchor #5 and for a linear indication on vane bank "E". The indications identified were characterized, sized and entered into the corrective action program for engineering evaluation and disposition. The inspectors examined the licensee's evaluation and disposition for continued operation without repair or rework of the linear indications by review of CR-PNP-2007-01590 (linear indication in vane bank "E") and CR-PNP-2007-01664 (linear indication in divider plate anchor #5).

CR-PNP-2007-01697 was initiated as a result of the identification of a linear indication found during the ultrasonic test of the RPV recirculation inlet nozzle (N2K) to safe end weld. The flaw was identified using an automated ultrasonic test technique that meets the requirements of the performance demonstration initiative's implementation of the ASME Code, Section XI, Appendix VIII, Supplement 10. The flaw was not easily characterized due to a lack of clear definition of the ultrasonic response due to previous repairs at the flaw location. Therefore, the licensee determined that a conservative estimate was appropriate and that a full structural weld overlay would be applied at this location. Also, in accordance with the guidance provided in NRC Generic Letter 88-01, Supplement 1, a sample expansion consisting of four additional welds of the same category were selected for examination. The inspectors reviewed test data, procedures and interviewed examiners and Level III nondestructive test oversight personnel participating in the repair activity. Also, the inspectors reviewed the ASME Section XI repair plan and observed the application of weld metal to a representative mock-up of the safe end to nozzle configuration. (See 4OA3)

b. Findings and Observations

No findings of significance were identified.

.5 Annual PI&R Sample Review - Core Shroud Tie Rod Repairs

a. Inspection Scope (1 sample)

The inspectors reviewed Entergy's actions, taken during RFO 16, to address General Electric (GE) Company's Part 21 Notification - BWR Core Shroud Tie Rod Upper Support Cracking. Cracking was discovered at one plant during an IVVI of the tie rod upper supports. The apparent root cause was Intergranular Stress Corrosion Cracking

(IGSCC) in the Alloy X-750 tie rod upper support material. This inspection comprised a review of Entergy and GE proprietary information related to the repair, related correspondence between NRC and Entergy regarding the repair, and review of video inspection records both prior and subsequent to the repairs.

b. Findings and Observations

No findings of significance were identified. Entergy submitted a request for authorization under the provisions of 10 CFR 50.55a(a)(3)(I) for modification of the core shroud stabilizer assemblies (tie rods). NRC approved Entergy's use of the GE proposed repair to replace the tie rod upper support with a modified upper support design capable of operation through the end of a renewed operating license term. Two of the four supports were replaced, however, due to difficulties encountered while attempting to replace the third support and a lack of special tooling needed to continue, Entergy decided to delay the replacement of the remaining two tie rods until RFO 17. Operation with two new and two old tie rod supports was evaluated and found acceptable. Inspection of the accessible surfaces of the supports that were not replaced found no indication of IGSCC.

4OA3 Event Follow-up (71153)

.1 Missed Reactor Protection System (RPS) Technical Specification Surveillance

a. Inspection Scope (1 sample)

On June 22, 2007, a “B” RPS relay failed. Entergy replaced the relay and, during review of PMT requirements, identified that scram contactor time response testing would be required. On June 25, 2007, the inspectors were informed of the identification of a missed TS surveillance requirement to perform time response testing on RPS scram contactors. Entergy evaluated operability of the RPS system and determined that the system remained operable and that TS 4.0.3, “Surveillance Requirement Applicability,” would allow a delay period up to the limit of the specified surveillance frequency. Entergy conducted a risk evaluation of the missed RPS surveillance, in accordance with TS 4.0.3, and determined that the surveillance could be delayed up to 90 days with no significant increase in risk. The inspectors reviewed applicable surveillance TS requirements, RPS system drawings, operability and reportability requirements, the risk evaluation, and the adequacy of operator actions.

b. Findings

The inspectors questioned Entergy regarding the applicability of TS 4.0.3 given that the time response test had never been performed on four of the RPS scram contactors, as compared to missing a surveillance test following satisfactory initial system baseline testing that originally showed system operability. Entergy identified that industry guidance had been issued on this question in the form of Technical Specification Task Force (TSTF)-IG-06-01 dated May 2006. In the TSTF, the following question is asked, “If it is discovered that a Surveillance has never been performed or has never been completely performed, can SR 3.0.3 be used?” [Note: SR 3.0.3 is the standard TS equivalent of Pilgrim TS 4.0.3] The TSTF response states, “Yes, SR 3.0.3 applies in conditions in which a Surveillance has never been performed or has never been performed completely provided that there is an expectation that the Surveillance will pass when it is performed and that the associated system is OPERABLE.” An Unresolved Item (URI) will track NRC evaluation of this industry guidance to determine if additional NRC guidance is necessary to specify when TS 4.0.3 applies in the case of a missed surveillance where it is determined that the surveillance was never originally performed to establish initial system operability. **(URI 05000293/2007003-04, Application of TS 4.0.3 When it is Discovered that a Surveillance Has Never Been Performed)**

Entergy modified the applicable surveillance procedures and successfully response time response tested all RPS scram contactors.

.2 LER Review and Closeout (2 samples)

(Closed) LER 05000293/2007-01-00, Primary Containment Isolations Following a Manual Reactor Scram.

The inspectors reviewed Entergy's actions associated with LER 50-293/2007-01-00. The LER provided an accurate description of the event, and follow-up actions, taken or planned, were appropriate to address the event. The event and associated corrective actions were captured in Entergy's corrective action program as CR-PNP-2007-00949, CR-PNP-2007-00939, and CR-PNP-2007-00934. No findings of significance were identified. The LER discusses a procedure adherence issue when resetting the isolation signal. The inspectors determined that this failure to comply with TS 5.4.1 constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's enforcement policy. This LER is closed.

(Closed) LER 05000293/2007-03-00, Reactor Coolant Pressure Boundary Leakage Due to Reactor Vessel Nozzle Weld Crack Propagation.

The inspectors reviewed Entergy's actions associated with LER 50-293/2007-03-00. The LER provided an accurate description of the event and follow-up actions, taken or planned, were appropriate to address the event. No findings of significance were identified and no violation of NRC requirements occurred. This LER is closed.

4OA6 Meetings, Including Exit

On July 17, 2007, the inspection results were presented to Mr. K. Bronson and other members of his staff. Proprietary information was reviewed for noble metals injection and the core shroud tie rod repair. No proprietary information was identified within this report.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by Entergy and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Technical Specification 5.4.1, "Procedures," requires in part that Entergy establish and implement procedures recommended in RG 1.33, Revision 2, Appendix A. Procedure 3.M.2-12.4, "Backfilling Reference Lines for Racks C2205, C2275, and C2251 Instruments (11, 12A, and 13A Condensing Chambers)" was developed to support startup from shutdown by ensuring all instruments and the associated active reference legs have been filled/flushed. On May 7, 2007, contrary to the above, I&C personnel failed to properly implement procedure 3.M.2-12.4 and left PT-263-51A isolated contrary to the procedure's restoration instructions which directed the PT's instrument rack isolation valve be open/verified open. The isolated pressure detector was identified during plant startup when it failed to respond to corresponding increases in reactor pressure. The finding was of very low safety significance (Green) because the high pressure reactor trip safety function remained operable due to redundant pressure detectors. The finding was more than minor because it adversely impacted the configuration control attribute of the reactor safety Mitigating Systems cornerstone and the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Entergy documented this issue in the corrective action program in CR-PNP-2007-02468.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

K. Bronson	Site Vice President, Pilgrim
B. Cobb	Maintenance Supervisor
T. Collis	System Engineer
J. Fitzsimmons	Radiation Protection Supervisor
B. Ford	Licensing Manager
G. James	Reactor Engineering Superintendent
J. Keenan	System Engineer
T. McElhinney	Chemistry Superintendent
J. Moylan	Electrical Supervisor
D. Noyes	Operations Manager
M. Santiago	Training Superintendent
R. Smith	General Manager-Plant Operations

NRC personnel:
W. Raymond, Senior Resident Inspector
C. Welch, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000293/2007003-04	URI	Application of TS 4.0.3 When it is Discovered that a Surveillance Has Never Been Performed (Section 4OA3.1)

Opened and Closed

05000293/2007003-01	NCV	Failure to provide adequate instructions for adjusting MO-1201-85 packing resulted in premature packing failure. (Section 1R12)
05000293/2007003-02	NCV	Failure to follow procedures resulted in the loss of shutdown cooling. (Section 1R13)
05000293/2007003-03	NCV	Inadvertent Decrease in Reactor Vessel Level Due to Personnel Error. (Section 1R20)

Closed

05000293/2007-01-00	LER	Primary Containment Isolations Following a Manual Reactor Scram. (Section 4OA3.2)
05000293/2007-03-00	LER	Reactor Coolant Pressure Boundary Leakage Due to Reactor Vessel Nozzle Weld Crack Propagation. (Section 4OA3.2)

LIST OF DOCUMENTS REVIEWED

References for Section 1R04
2.2.85.1, Augmented Fuel Pool Cooling (With Shutdown Cooling) Mode 1, Revision 11
P&ID M231, Fuel Pool Cooling and Demineralizer Revision 37
P&ID M241, residual Heat Removal System Revision 7
2.1.12.1 R62, Emergency Diesel Generator Surveillance
2.2.8 R90, Standby AC Power System (Diesel Generator)
2.2.108 R40, Diesel Generator Cooling and Ventilation System

References for Section 1R05
Condition report 200701785, 200701829, 200702034
EN-DC-127, Control of Hot Work and Ignition Sources, Revision 2
ENN-DC-161, Transient Combustible Program Revision 1
5.5.2, Special Fire Procedure Revision 36
8.B.17.2, Inspection of Fire Damper Assemblies, Revision 10
89Xm-1-ER-Q, Updated Fire Hazards Analysis, Revision E7

References for Section 1R06
Condition Report 200701513

References for Section 1R07
8.5.3.14.2, "RHR Heat Exchanger Thermal Performance Test," Revision 2
Calculation M-170, "RHR Heat Exchanger Allowable Fouling Resistance," Revision 2

References for Section 1R08

NDT Examination Reports

07-M-339	Magnetic particle examination report, HPCI system, HE-26-175HL1(1)
07-P-383	Liquid penetrant test report, reactor recirculation system, 2R-N1B-14HL2(4)
07-U-360	Ultrasonic test report, core spray, weld 14-B-18
07-U-361	Ultrasonic test report, core spray, weld 14-B-19
APR-04	PT and UT examination summary sheet for weld 2R-N2A-1
APR-05	PT and UT examination summary sheet for weld 2R-N2B-1
APR-06	PT and UT examination summary sheet for weld 2R-N2C-1
PIL-R16-07-017	PT and UT examination summary sheet for weld RPV-N9B-1

Calibration Data Sheets

UT-038, 039 UT calibration and examination record for procedure TP04-032

NDT Examination Procedures

ENN-NDE-9.04 R1 - Ultrasonic examination of ferritic piping welds (ASME Section XI)

ENN-NDE-9.23 R0 - Ultrasonic examination of austenitic piping welds (ASME Section XI)
ENN-NDE-9.31 R0 - Magnetic Particle Examination (MT) for ASME Section XI
ENN-NDE-1.00 R0 - Personnel documentation and certification review report
ENN-EP-S-001 R0 - IWE General visual containment inspection personnel qualification
ENN-NDE-9.41 R0 - Liquid Penetrant Examination (PT) for ASME Section XI

In Vessel Remote Visual Examination
TP07-006 R0 - Invessel Visual Inspection (IVVI) of BWR 3 RPV Internals

Repair-Replacement
MR 07106245 - Replace 1-HO-64 MSL low point drain valve AO-203-1D
MR 07105779 R1- Restore disc seating surface on valve MO-1001-7C

Condition Reports
CR-PNP-2007-02035 - Weld data sheet not revised to reflect welding procedure (WPS) change
CR-PNP-2007-02051 - Documentation of welder qualification to weld on valve disc
CR-PNP-2007-02033 - Change of specified welding procedure for repair on MO-1001-7C disc
CR-PNP-2007-01432 - Divider Plate between bank C and D is distorted
CR-PNP-2007-01664 - Linear indication in divider plate anchor
CR-PNP-2007-01494 - Foreign material on top of steam dryer bank E
CR-PNP-2007-01590 - Linear indication of vane bank E on steam dryer ID side
CR-PNP-2007-01383 - Weld is undersize and root NDT was not performed
CR-PNP-2007-01777 - Interpretation of test results of the ultrasonic test of nozzle N2K
CR-PNP-2007-01697 - Scheduled examination of N2K nozzle safe end weld detected an indication of rejectable size in the inconel weld material

Corrective Actions
CR-PNP-2007-01697 - CA#1 Evaluation of the nozzle condition by engineering before assembly
CR-PNP-2007-01697 - CA#2 Scope expansion and weld overlay is required
CR-PNP-2007-01697 - CA#3 Define scope expansion and verify examinations are completed
CR-PNP-2007-01697 - CA#4 Verify the weld overlay of N2K is complete
CR-PNP-2007-01697 - CA#5 Perform apparent cause evaluation

Welding Procedures
TP07-051 Welding Procedure Specification WSI/WPS-03-08-T-804-103944
TP07-050 Welding Procedure Specification WSI/WPS-08-43-S-001
CS-1/1-B R2 Weld Procedure Specification, P1/P1, manual gas tungsten arc

Drawings
M1B51 R. EO - Reactor modification shroud repair
ISI-1-23-4 R5 - High pressure coolant injection system
ISI-1-14-1 E6 - Core spray system
ISI-1-2R-A - Reactor Recirculation System

Miscellaneous
DRF 0067-1649 Evaluation of indications on steam dryer unit end plate
DRF 0067-1650 Review and evaluation of steam dryer indications Pilgrim R16
BWRVIP-75-ABWR vessel and internals project technical report
CR–PNP-2007-02051 Operability of RHR valve MO-1001-7C

M-1245 - Calculation M-1245, Weld overlay plan for N2 and N9 nozzle to safe end welds
INR P116-IVVI-07-15 Linear indication on steam dryer divider plate anchor number 5
ASME IX Welder Qualification Logs

References for Section 1R11
O-RQ-04-04-54, (Revision 0) Instruction Module Reactor Startup and Criticality - (& Main Turbine Overspeed) Just In Time Training

References for Section 1R12
CR 20070949
MR 03117384
3.M.4-10 Rev. 31, Valve Maintenance
RWCU Maintenance Rule SSC Basis Document Revision 1
RWCU historical data

References for Section 1R13
Condition Report 200701503
8.M.2-8.1, "Calibration of ATS Transmitters Rack C2205," Revision 18
8.M.1-30, "ATWS System Calibration Test," Revision 50
2.4.25, "Loss of Shutdown Cooling," Revision 29
Computer Trended Data for Fuel Pool and RHR Temperatures
Station Operating Logs
RFO 16 Shutdown Risk Review Report
1.5.22, Risk Assessment Process, Revision 10
3.M.1-45, Outage Shutdown Risk Assessment, Revision 5
1.3.34.15, Protected Area Postings, Revision 0
NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management Dec. 1991

References for Section 1R15
M897 Revision 3, Pilgrim Nuclear Plant : ECCS Strainer Performance Analysis
CR200701277
Calculation M572, Revision 23; Torus Water Inventory Containment Isolation Valve Leak Rate
Calculation PNPS-1ERHS-XIII.BB-68, Revision 0; Offsite Radiological Consequences Due to a Loss of Collant Accident at 102% of 1998 MWT
CR200700719

References for Section 1R19

3.M.4.-8, Main Steam Isolation Valve Maintenance, Revision 42
3.M.4-8.1, Main Steam Isolation Valve Preventive Maintenance, Revision 9
8.7.1.6, Local Leak Rate Testing of the Main Steam Isolation Valves, Revision 24
8.I.11.21, Main Steam Isolation Valve Cold Shutdown Operability, Revision 2
8.7.4.4, Main Steam Isolation Valve Operability at 60% Power, Revision 2
8.M.1-15, MSIV Limit Switches, Inspection, Adjustment, and Functional Test, Revision 22
VT-1 Examination Report No. VT-1-07425
VT-2 Examination Report No. VT-54-07045
VT-1 Examination Report No. VT-1-07424
VT-2 Examination Report No. VT-54-07046
VT-3 Examination Report No. VT-2-07427
8.Q.3-8.2, Limitorque Type HBC SB/SMB-0 through SB/SMB-3 Valve Operator Maintenance, Revision 12
3.M.4-10, Valve Maintenance, Revision 36
8.7.1.2, Torus Water Inventory Primary Containment Isolation Valves Leak Rate Test, Revision 9
8.7.1.6, Local Leak Rate Testing of the Main Steam Isolation Valves, Revision 24
CR 200701340, 200701363
8.5.2.3, LPCI and Containment Spray Motor-Operated Valve Operability Test, Revision 47

References for Section 1R20

RFO 16 Shutdown Risk Review Report
Power Maneuvering Plan PMP-MAN.C16-86 Cycle 16 Shutdown
2.1.5, Controlled Shutdown from Power, Revision 101
2.2.19.1, Residual Heat Removal System - Shutdown Cooling Mode of Operation, Revision 22
2.4.25, Loss of Shutdown Cooling, Revision 29
Temporary Procedure TP07-022, RHR Loop B Shutdown Cooling Operations During NOBLECHEM Application Revision 0
2.2.85.1, Augmented Fuel Pool Cooling (With Shutdown Cooling) Mode 1, Revision 11
OPER-07, RPV metal Temperatures and Pressures, 4/18/05
4.3, Fuel Handling, Revision 109
OPER-13, Daily Refueling Checklist, Revision 109
OPER-10, Refueling Checklist, Revision 109
3.M.1-14, General Maintenance Procedure for Heavy Load Handling Operations, Revision 19
3.M.4-48.1, Opening And Closing of Reactor Pressure Vessel, Pre-outage Revision 9
3.M.4-48.2, Opening and Closing of Reactor Pressure Vessel, Disassembly, Revision 20
3.M.4-9, Inspection of the Drywell and Suppression Chamber, Revision 11
Engineering Change Notice 07105979, “Rigging Plan for 360 Degree Work Platform
Engineering Request 06116442 , Installation and Use of New GE 360 Degree Work Platform, Revision 0
2.1.8.5, Reactor Vessel Pressurization and Temperature Control for Class 1 System Leakage Test, Revision 19
2.1.1, Startup From Shutdown, Revision 158
2.1.4, approach to Critical, Revision 25
2.1.14, Station Power Changes, Revision 92
9.16.1, In-Sequence Critical for Shutdown Margin Demonstration, Revision 11
9.16.2, Subcritical demonstration Revision 7
9.9, Control Rod Scram Insertion Time Evaluation, Revision 60
Power Maneuvering Plan MAN.C-17-01, Cycle 17 Startup

Technical Specification 3.10.A and 4.10 A, Refueling Interlocks
Technical Specification 3.10.B, Core Monitoring
UFSAR Section 7.5.4, Source Range Monitoring
UFSAR Section 7.6, Refueling Interlocks
Temporary Procedure TP06-030, “Administrative Controls for Recirculation Pump Motor Project Heavy Load Handling Activities in the Drywell,” Revision 0
Calculation C15.0.3433, “Drywell Structures for P201B Recirculation Pump Motor Replacement,” Revision 2
Engineering Request 04107154, “Recirculation Pump Motor Service Platform,” Revision 0
Engineering Request 05111500, “Rigging Plan for Recirculation Pump Project,” Revision 0
Engineering Request 07104780, Figure P3, “Bridging/Spacers Beams at 23 ft Elevation
License Amendment No. 227, Extension of Pressure-temperature Limits Specified in Technical Specifications, 3/26/07
In vessel Inspection Report INR-P1R16-IVVI-07-15, Steam Dryer divider Plate, 4/16/07
Maintenance Requests (MR) 05108348, 07105830, 05110469, 05109060, 03121989, 03121990
Condition Reports 200701257, 200701199, 200701229, 200701361, 200701437, 200701466, 200701432, 200701494, 200701495, 200701497, 200701518, 200701520, 200701590, 200701664, 200701666, 200701710, 200701741, 200701738, 200701752, 200701700, 200701710, 200701741, 200701821, 200702056

References for Section 1R22
8.M.3-1, “Special Test for Automatic ECCS Load Sequence of Diesels and Shutdown Transformer with Simulated Loss of Offsite Power and Special Shutdown Transformer Load Test;” Revision 50
Technical Specification Surveillance Requirement 4.9.A.1
Condition Reports 200702020, 200702021, 200702022, 200702016
NRC Report 2006-05 Finding NCV 0500293/2006005-01: Failure to perform an adequate 50.59 evaluation for a change to a surveillance test required by Technical Specification 4.9.A.1.

References for Section 1R23
Temporary Modification-06-01-056, Temporary alteration for Installation of Noble Metals Sample Equipment
Temporary Procedure TP06-028, Special Test for Noble Metals Chemical Application Procedure, Revision 0
General Electric (GE) Energy Report GE-NE-0000-0050-0979-01-R0, Assessment Report Pilgrim Nuclear Power Station NobleChem Application and Monitoring, Revision 0
GE Energy Report GE-NE-0000-0050-0979-02-R0, Noble Metal Chemical Addition Technical Safety Evaluation for Pilgrim Nuclear Power Station, Revision 0
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Drawing M252, Nuclear Boiler, sheet 2, Revision E59
InVessel Visual Inspection (IVVI) Final Report #1955-PNPS-07-MJZMD, dated May 2007.

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TS 3.1, Reactor Protection System (RPS)
TS 4.03, Surveillance Requirement Applicability, and Bases
RPS Trip Channel Relays and Associated Scram Contactor Actuator Logics Diagram
Missed SR 4.1.2 Surveillance for RPS Risk Evaluation
CR-PNP-2007-03060, Relay 5a-K14F not Time Response Tested
CR-PNP-2007-03072, Relays 5A-K14E/F/G/H not Time Response Tested

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ALARA	as low as reasonable achievable
ASME	American Society of Mechanical Engineers
ATS	analog trip system
CEDE	committed effective dose equivalent
CFR	code of federal regulations
CR	condition report
ECCS	emergency core cooling system
EDG	emergency diesel generator
FME	foreign material exclusion
GE	General Electric
gpm	gallons per minute
HPCI	high pressure coolant injection
I&C	instrumentation and controls
IMC	inspection manual chapter
IR	inspection report
ISI	inservice inspection
IGSCC	intergranular stress corrosion cracking
IVVI	in vessel visual inspection
JIT	just in time
LER	licensee event report
LLTR	local leak rate tests
mrem	millerem
MR	maintenance request
NCV	non-cited violation
NDE	non-destructive examination
NDT	non-destructive test
NRC	Nuclear Regulatory Commission
OA	other activities
PARS	Publically Available Records System
PCIS	primary containment isolation system
PDI	performance demonstration initiative
PI	performance indicator
PI&R	problem identification and resolution
PM	preventive maintenance

PMT	post-maintenance test
PNPS	Pilgrim Nuclear Power Station
PT	pressure transmitter
RBCCW	reactor building closed cooling water
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RFO	refueling outage
RG	regulatory guide
RHR	residual heat removal
RPS	reactor protection system
RPV	reactor pressure vessel
RWCU	reactor water cleanup
RWP	radiation work permits
SDP	significant determination process
SFP	spent fuel pool
SRO	senior reactor operator
SSC	system, structure, or component
TS	technical specification
UFSAR	updated final analysis report
URI	unresolved item
UT	ultrasonic test
VT	visual test