July 30, 2003

Mr. Michael Balduzzi
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 05000293/2003006

Dear Mr. Balduzzi:

On June 28, 2003, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Pilgrim reactor facility. The enclosed integrated inspection report documents the inspection findings, which were discussed on July 2, 2003, 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.

The report documents three self-revealing findings of very low safety significance (Green), which involved violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI of the NRC Enforcement Policy. If you contest these non-cited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region 1; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-001; and the NRC Resident Inspector at the Pilgrim facility.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial power nuclear power plants during calender year '02 and the remaining inspection activities for the Pilgrim Nuclear Power Station are scheduled for completion in July 2003. The NRC will continue to monitor overall safeguards and security controls at the Pilgrim Nuclear Power Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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/

Clifford Anderson, Chief
Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 05000293/2003006
w/Attachment: Supplemental Information

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	05000293/2003006
Licensee:	Entergy Nuclear Generation Company
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	March 30, 2003 - June 28, 2003
Inspectors:	W. Raymond, Senior Resident Inspector C. Welch, Resident Inspector T. Burns, Reactor Inspector J. Furia, Senior Health Physicist C. Cahill, Senior Reactor Inspector J. Talieri, Reactor Engineer R. Cooney, Contractor A. Blamey, Senior Operations Engineer (in-office) A. Lohmeier, Senior Reactor Engineer (in-office)
Approved By:	Clifford Anderson, Chief Projects Branch 5 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000293/2003-006; 03/30 - 06/28/03; Pilgrim Nuclear Power Station, Resident Inspection. Personnel Performance During Non-routine Plant Evolutions, Refueling and Other Outage Activities, and Surveillance Testing.

The report covered a 13-week period of inspection by resident and region-based inspectors. Three Green non-cited violations 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). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Self-Revealing Findings

Cornerstone: Initiating Events

Green. An operator manipulated the incorrect main control board (MCB) switch, which resulted in the automatic closure of the main steam isolation valves and shutdown of the reactor on May 19, 2002. The inspector identified a non-cited violation of Technical Specification 5.4.1.a because the operator failed to properly implement procedure 2.1.1, "Startup from Shutdown," by failing to properly operate the pressure regulating system, maintain the required MPR setpoint, and heed the procedure caution. The finding is more than minor because it led to a plant trip. This human performance error was determined by a phase 3 risk analysis to be of very low safety significance because the reactor decay heat was low, the operators could recover the main condenser as the normal heat sink, and mitigating systems were available following the shutdown. (Section 1R14)

Green. An inadequate tagout restoration resulted in an unintended drain path from the reactor vessel on April 27, 2003. The inspector identified a non-cited violation of Technical Specification 5.4.1.a because the operators failed to properly implement Section 6.2.10 of procedure 1.4.5, "PNPS Tagging Procedure," by failing to ensure appropriate restoration positions and sequences were specified. The loss of reactor water level is an issue that is more than minor because it is a precursor to a more significant event, the loss of shutdown cooling. The finding did not degrade the licensee's ability to terminate the leak path or recover decay heat removal, if lost. Because the loss of level was less than 24 inches, the finding was determined to be of very low safety significance (Green) when assessed in accordance with MC 0609, Appendix G. (Section 1R20)

Green. A non-cited violation of Technical Specification 5.4.1.a was identified because a test procedure for testing emergency power sources was not adequate. This resulted in the inadvertent drain down of the reactor vessel through the automatic depressurization system (ADS) valves to the torus on May 9, 2003. The procedure failed to establish initial plant conditions or conditions to inhibit the ADS to prevent a drain down of the reactor vessel through the ADS valves. The loss of reactor water level is an issue that is more than minor because it is a precursor to a more significant event, the loss of shutdown cooling. The finding did not degrade the licensee's ability to terminate the leak path, recover decay heat removal, if lost, or impact the ability to establish a heat removal path to the suppression pool. Because the loss of level was less than 24 inches, the finding was determined to be of very low safety significance (Green) when assessed in accordance with MC 0609, Appendix G. (Section 1R22)

B. <u>Licensee Identified Violations</u>

Two violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned have been entered into the licensee's corrective action program. The violations are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station began the period operating at full power and shutdown on April 18, 2003 to conduct refueling outage RFO #14. The unit was removed from the electrical grid at 3:02 a.m. on April 19 and the reactor entered cold shutdown at 5:59 p.m. on April 19, 2003. Major outage maintenance activities included replacing 164 fuel bundles and 32 control rod drive mechanisms. The plant design changes during the outage included the installation of the reactor vessel level modification, and the changes to support the increase in licensed power level as part of the thermal power optimization project.

The reactor was taken critical at 6:41 a.m. on May 12 and the plant was synchronized with the electrical grid on May 13. Power ascension was limited to 35% full power as the licensee evaluated increases in drywell leakage. The reactor was shutdown on May 15 and the plant taken to cold shutdown on May 16 to repair the A and B recirculation pump canopy seal welds and other sources of drywell leakage. The plant was restarted with the reactor taken critical at 22:17 p.m. on May 18.

The reactor automatically shutdown from 2% full power at 4:24 a.m. on May 19 when the main steam isolation valves closed while pressurizing the plant in preparation for rolling the main turbine. The post-trip review indicated all plant safety systems responded as expected. After completing the post-trip evaluation, the plant was restarted on May 20 with the reactor taken critical at 02:49 a.m., and operation at full power resumed on May 22.

The reactor automatically shutdown from 98.5% full power at 8:50 a.m. on June 1 in response to a full load reject following a fault on the unit auxiliary transformer. The post-trip review indicated all plant safety systems responded as expected. After completing an evaluation and temporary modification to operate the plant with internal loads on the startup transformer, the plant was restarted at 17:07 p.m. on June 2, the reactor taken critical at 22:45 p.m., the generator was placed on line on June 3. The plant power was held at 48% on June 3-4, while a condenser tube was plugged in the 1-1 water box. Plant operation at full power resumed on June 4. The plant began operations at the new thermal power limit of 2028 MWth on June 9, 2003 (License Amendment 201).

The plant subsequently operated at full power except for brief periods of operation at reduced power for routine condenser maintenance, testing and rod pattern adjustments.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity)

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The inspector reviewed seven selected safety evaluations associated with initiating event, mitigating, and barrier integrity cornerstones to verify that changes to the facility or procedures as described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59, and that the safety issues pertinent to the changes were properly resolved or adequately addressed. These safety evaluations were selected based on the safety significance of the changes and the risk to structures, systems and components.

The inspectors also reviewed fifteen screen-out evaluations for changes, tests and experiments for which the licensee determined that safety evaluations were not required. This review was performed to verify that the licensee's threshold for performing safety evaluations was consistent with 10 CFR 50.59.

In addition, the inspectors reviewed the administrative procedure that was used to control the screening, preparation, and issuance of the safety evaluations to ensure that the procedure adequately covered the requirements of 10 CFR 50.59.

The listing of the safety evaluations and screen-out evaluations reviewed is provided in the attachment to this report.

c. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed partial system walkdowns of the residual heat removal (RHR) system while aligned for shutdown cooling and alternate fuel pool cooling. These inspections verified that key valves and breakers were properly positioned, that critical components were tagged to restrict operation as required by procedure 2.2.19.1, "Residual Heat Removal System - Shutdown Cooling Mode of Operation," and that system parameters of flow and temperature were within the required band and provided positive indication that decay heat was in fact being adequately removed from the reactor core and spent fuel pool. The inspectors reviewed appropriate system drawings and procedures to determine the correct system lineup. Walkdowns were performed on:

- April 21 - while in shutdown cooling on the B RHR loop
- April 25 - while in alternate fuel pool cooling mode 1 on the B RHR loop
- May 2 - while in alternate fuel pool cooling mode 1 on the A RHR loop

The inspector conducted a partial system review of the high pressure coolant injection (HPCI) system during the time when the reactor core injection cooling (RCIC) system was out of service for scheduled testing and then corrective maintenance. The inspector reviewed the appropriate system drawings (M243 and M244 for HPCI) and valve line-up procedures to verify the correct system lineup. The Updated Final Safety Analysis Report and the Technical Specifications were reviewed to ascertain the required system configuration.

- HPCI System review - April 17 for CM on RCIC PS-1360-9C

The inspector conducted a partial system review of the control rod drive (CRD) system during the time fuel shuffle and control rod blade removal was in progress. The inspector reviewed the appropriate system drawings (M250 for CRD) and valve line-up procedures to verify the correct system lineup.

- CRD System review per 2.2.87.4 and Tagouts 2003097 and 2003098.

A complete system walkdown of the accessible portions of the RHR system was performed to evaluate the system's capability to perform its safety function. The position of critical valves, breakers and switches were verified in the proper position and the material condition of the system was assessed based on visual inspection of the accessible areas and a review of condition reports, surveillance test data, maintenance rule information, maintenance work requests, and discussion with the system engineer. The updated final safety analysis report, system drawings and procedures, and the technical specifications were used in support of the inspection.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Quarterly Fire Protection Inspection

a. Inspection Scope

The inspectors toured selected areas of the plant to observe conditions related to: (1) transient combustibles and ignition sources; (2) the material condition and readiness of fire protection systems and equipment; and (3) the condition and status of readiness of fire barriers used to prevent fire damage or fire propagation. The inspector verified that any identified degraded conditions were compensated by compensatory measures until appropriate corrective actions could be taken. The inspector also reviewed the applicable fire hazard analysis fire zone data sheets and selective surveillance procedures to ensure that the specified fire suppression systems surveillance criteria were met. The areas inspected included:

- Fire Zone 1.30A, Torus Compartment
- Fire Zone 1.30, Drywell
- Fire Zone 1.21, A RBCCW Pumps/Heat Exchanger Room
- Fire Zone 1.22, B RBCCW Pumps/Heat Exchanger Room
- Fire Zone 1.13, Reactor Building - Fuel Pool Cooling and Heat Exchangers
- Fire Zones 4.1 & 4.3, A & B EDGs

Additionally, the inspector reviewed the completed test results for the fire pumps obtained per 8.B.1, “Fire Pump Test,” and verified that proper hot work controls were in place for ongoing hot work in the RBCCW rooms.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspector conducted a walkdown of the reactor building and auxiliary heat exchanger rooms to assess the effectiveness of internal flood control measures. The references used for this review are listed in the attachment to this report. The inspector reviewed control room panels and alarms, and interviewed licensed operators. Special emphasis was placed on the flooding controls for the core spray and residual heat removal systems due to the risk significance of those systems.

Items selected for review during the walkdowns included watertight piping penetrations, watertight doors, floor level alarms, and floor sump systems. Passive equipment such as curbing and drains were inspected. The troughs and gratings in the auxiliary bay floor, which discharges into the torus room via a loop seal, was inspected and found free of debris. The inspector compared licensee procedure controls with those described in the internal flood analysis in the Updated Final Safety Analysis Report Section 10.7.6.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspector reviewed the thermal performance test results for the B Residual Heat Removal and the B Reactor Building Closed Cooling Water (RBCCW) heat exchangers to verify that the performance monitoring techniques used to ensure heat removal capabilities were acceptable. The testing was completed on April 19, 2003, per 8.5.3.14.1, “RBCCW Heat Exchanger Thermal Performance Test,” and per 8.5.3.14.2, “RHR Heat Exchanger Thermal Performance Test.” The inspector used Calculations M-710, “Heat Exchanger Performance Testing,” and M-641, “RBCCW Heat Exchanger Performance” as references for this review. The inspector verified that the inspection results were compared against established acceptance criteria; the performance monitoring considered the differences between plant conditions and design conditions; the frequency of testing and inspections was sufficient; and, the licensee had a program for bio-fouling control. The inspector verified that the results were evaluated to ensure proper heat exchanger operation, and discrepancies were evaluated and corrected. The inspector reviewed a sample of corrective action condition reports related to the selected equipment to verify that identified problems were appropriately resolved. The inspector conducted a walkdown of the heat exchanger to assess material conditions and observed licensee maintenance activities during the outage to clean, inspect and repair the RBCCW heat exchanger.

The inspector reviewed the following heat exchangers to verify that they remained capable of fulfilling their required safety functions.

- B Residual Heat Removal (RHR)
- B Reactor Building Closed Cooling Water (RBCCW)

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI)

a. Inspection Scope

The inspector observed selected samples of in-process nondestructive examination (NDE) activities. Also, the inspector reviewed documentation of additional samples of NDE and repair/replacement activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation could result in a significant increase in risk of core damage. The observations and documentation review was performed to verify activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspector reviewed a sample of inspection reports initiated to document the performance and record results of ISI examinations completed during this period. Also, the inspector evaluated the licensee's effectiveness in resolving relevant indications identified during ISI activities.

The inspector observed manual ultrasonic testing (UT) and reviewed selected documentation of penetrant test (PT) activities to verify the effectiveness of the examiner and process for identifying degradation of risk significant systems, structures and components. The inspector evaluated the activities for compliance with the requirements of ASME Section XI of the ASME Boiler and Pressure Vessel Code. The inspector examined the licensee's evaluation and disposition for continued operation without repair or rework of non-conforming conditions identified during ISI activities by review of nonconformance report 97-124 and condition report (CR) 2003-01472 for indications observed during visual examination of the steam dryer leveling screws.

The inspector observed the UT testing performed on residual heat removal (RHR) and high pressure core injection (HPCI) field welds (FW) DC-10-F9 and HL-23-F22, 18 and

16 inch diameter pipe to valve and pipe to pipe butt welds, respectively. In addition, the inspector reviewed documentation of the PT of RHR FW DC-10-F9 and the magnetic particle test (MT) of HPCI FW HL-23-F22. The inspector reviewed a portion of the remote in-vessel visual inspection of the reactor steam dryer base metal, structural welds and leveling bolts. The review was conducted 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.

The inspector reviewed welding activities associated with the replacement of selected components to verify the activities were performed in accordance with the requirements of ASME Sections IX and XI. The inspector reviewed selected portions of maintenance request (MR) 01110262 for the replacement of the bonnet drain valve on the shutdown cooling suction valve (MO-1001-50) in the RHR system. The inspector reviewed the drain valve substitution equivalency evaluation (SEE 1064) and the shop fabrication and field installation of the valve and associated piping assembly. Also, the inspector reviewed MRs 02119650 and 02119656 used for the fabrication of salt service water pipe spools JF29-4-9, 5-4, 6-1, 9-6 and 9-7 to verify the activities were in accordance with the applicable ASME Code requirements. The inspector reviewed the work instructions, welding instructions, welding procedure specification P1-TS, revision 28, weld procedure qualification record YA-PQR-39-0 (5/5/1999), and the NDE requirements and test results of the completed welds.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Training

a. Inspection Scope

The inspector observed training activities for licensed senior reactor operators and reactor operator candidates conducted on June 5, 2003 as part of the licensed operator initial training program. The purpose of the inspection was to review the adequacy of licensee procedures and training for high-risk operator actions associated with mitigating postulated anticipated transient without scram scenarios. The review included a consideration of plant operating experiences with control rod position indication following a scram. The references used in the review are identified in the attachment to this report. The inspector assessed the methods used to assure timely implementation of emergency operating procedure EOP-2 following a failure to scram. The inspector verified that licensee procedures and operator training assured timely actions are taken to control power and vessel pressure and level, and initiate the standby liquid control system while rod positions are verified and rods are inserted via alternate means.

The inspector observed the operator performance during the conduct of testing on the simulator on June 9, 2003 as part of the licensed operator requalification program. The training was performed to demonstrate the operator's ability to implement alarm and emergency operating procedures without the use of control room annunciators. The scenarios involved abnormal operational transients with multiple problems and equipment failures. The inspector verified that the crew met the training scenario objectives and performed the critical tasks. The inspector verified proper use of the system operating procedures and emergency operating procedures to stabilize the plant. The inspector also verified that the post-scenario critique discussed any relevant lessons learned.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule

a. Inspection Scope

The inspector reviewed the implementation of the maintenance rule (10 CFR 50.65) as applied to the residual heat removal and control rod drive (CRD) systems. This inspection comprised review of applicable maintenance rule basis and tracking documents, the CRD and RHR system report card, the Updated Final Safety Analysis Report, and interviews with the RHR system engineer and maintenance rule coordinator. The following equipment issues were reviewed:

- CR 200209070, motor operated valve MO-1001-28A failure
- CR 200300642, motor operated valve MO-1001-29A failure
- CR 01.09004, Bus B6 Inoperable Due to Incorrect Relays (LER 2001-01)
- CR 03.2152 and 2155, CRD 03-35 Inoperable Following Replacement

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspector evaluated on-line risk management for planned and emergent work. The inspector reviewed maintenance risk evaluations, work schedules, recent corrective actions, and control room logs to verify that other concurrent planned and emergent maintenance or surveillance activities did not adversely affect the plant risk already incurred with the out of service components. The inspector verified that the licensee took the necessary steps to control work activities, took actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems. The inspector assessed Pilgrim’s risk management actions during plant walkdowns. The inspector also discussed the risk management with maintenance, engineering and operations personnel for the following activities:

- MR 03105244, LPCI Loop A Injection Valve #1 Inspection (CR 200300642)
- 3.M.4-1, Control Rod Drive Removal and Installation
- MR 03110225, B H2O2 Analyzer (LCO 1-03-0172)
- Leak Rate Testing on Shutdown Cooling Isolation Valves MO-1001-47 and 50.
- Planned Maintenance on the B train of Standby Gas Treatment

The inspector reviewed the contingency actions established for securing shutdown cooling on Saturday April 26, to accomplish local leak rate testing of the shutdown cooling isolation valves. This inspection included review of the drain plan, the draft fill plan and its proposed changes, observation of the pre-evolution brief, and discussion with the operating crew regarding the abort criteria for securing the test and initiating actions to refill and vent the RHR system to restore decay heat removal.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

Condenser Tube Leakage

The inspector reviewed the operator response to a condenser tube leak during operations at 40% full power on June 3, 2003, and the actions to maneuver the plant and to isolate the condenser water boxes (reference condition report 200302315). This inspection focused on whether the response to the off normal condition was in accordance with station procedure 2.4.33, “Condenser Chloride Intrusion,” and technical specification requirements. The inspection included: a review of the technical specifications, logs, and plant parameters; discussions with operations and chemistry personnel; and a walkdown of the control room panels.

June 1 Reactor Scram due to Failure of the Unit Auxiliary Transformer

The reactor automatically shutdown from 98.5% full power at 8:50 a.m. on June 1 in response to a full load reject following a trip of the phase A differential relay due to a fault on the unit auxiliary transformer. The inspector reviewed the operator’s response for the automatic shutdown to determine if the response was in accordance with station procedures and training, if operator performance impacted the event, and to verify that the safety equipment functioned properly. This inspection consisted of a review of the post trip report, operator logs, plant computer data, the event report, and discussion with plant personnel.

May 19 Reactor Scram during Startup due to Operator Error

The reactor automatically shutdown from 2% full power at 4:24 a.m. on May 19 when the main steam isolation valves closed while pressurizing the plant in preparation for rolling the main turbine. The inspector reviewed the operator's response for the automatic reactor shutdown to determine if the response was in accordance with station procedures and training, if operator performance impacted the event, and to verify that the safety equipment functioned properly. This inspection consisted of a review of the post trip report, operator logs, plant computer data, the event report, and discussion with plant personnel. (4OA3 and 4OA4)

b. Findings

Introduction. Green. Operation of the incorrect main control board (MCB) switch resulted in the automatic shutdown of the reactor. This human performance error was of very low safety consequence (Green) and constituted a non-cited violation of Technical Specification 5.4, "Procedures."

Description. During the reactor plant startup on May 19, operation of the mechanical pressure regulator (MPR) was successfully verified in accordance with procedure 2.1.1, "Startup from Shutdown," at the 200 psig setpoint, with very little deviation between the indicated setpoint and reactor pressure. Subsequent to this initial verification, the procedure directed the operators to adjust the MPR setpoint as necessary to maintain it 40 to 80 psi above reactor plant pressure which was being increased to normal operating conditions. The procedure, at 525 psig, directed the operators to verify that the electrical pressure regulator was set to 940 psig and cautioned that the actual MPR setpoint could be slightly lower than the indicated MPR setpoint. At approximately 870-880 psig reactor pressure, MCB alarm "Bypass Valve Not Closed" unexpectedly energized and the #1 BPV indicated partially open. The operator immediately responded to the alarm and operated what he believed to be the control switch for the MPR setpoint with the intent of raising the pressure setpoint to close the B.V. At the time, the MPR was set to approximately 900 PSEG. A review of the event identified that the operator had mistakenly operated the adjacent bypass valve opening jack (BVOJ) control switch. This caused the #1 and #2 BPVs to fully open and the #3 valve to partially open. With the BPVs held open by the BVOJ, reactor plant pressure dropped and reactor level concurrently rose. The safety systems responded to the transient and per design closed the main steam isolation valves and automatically shutdown the reactor. The operators responded in accordance with station procedures and terminated the event and established normal shutdown conditions and decay heat removal.

Analysis. The operator error that resulted in the reactor scram with closure of the main steam isolation valves was considered greater than minor because it involved a procedural violation that resulted in an initiating event, which is similar to example 4.b of Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues." Phase 1 of the At-Power Reactor Safety Significance Determination Process (SDP) screened this finding to Phase 2 because it adversely impacted both the initiating events and mitigating systems cornerstones. Phase 2 estimated the risk significance of this finding due to internal initiating events as White (low to moderate safety significance). The assumptions made in the Phase 2 analysis were as follows.

The inspectors modified the Risk-Informed Inspection Notebook for Pilgrim Nuclear Power Station (SDP Phase 2 notebook) to reflect changes that were developed during the SDP benchmarking effort conducted the week of June 16, 2003. The modifications to Table 3.2, "SDP Worksheet for Pilgrim Nuclear Power Station, Unit 1 - Transients without PCS [Power Conversion System] (TICS)," included:

- Crediting the fire water pumps for late injection through the low pressure coolant injection cross-tie to the reactor pressure vessel;
- Revising the containment venting operator action credit to 3; and
- Revising the depressurization operator action credit to 3.

The inspectors assumed that the initiating event likelihood rating was 0 (Note: The initiating event likelihood rating = - $\log_{10}$(initiating event frequency)., because the operator error directly resulted in a reactor scram with closure of the main steam isolation valves.

Recovery credit for the power conversion system was assumed because sufficient time was available for the operators to manually recover the main condenser using Procedure 2.1.5, "Controlled Shutdown from Power," operators had been trained on this procedure in both the initial licensing and requalification training programs; environmental conditions did not adversely impact these recovery actions; and no special equipment was needed to perform these recovery actions.

However, a review of the Phase 2 results indicated that they were conservative for two reasons. First, the Phase 2 SDP only allows a recovery credit of 1, which was conservative by at least two orders of magnitude for this case. Second, the event occurred at low power during the startup following a refueling outage when decay heat was very low. Therefore, the inspectors determined that a Phase 3 analysis of this finding was appropriate.

The Phase 2 SDP framework was used for the Phase 3 analysis because it identified the appropriate dominant accident sequences. The Phase 3 analysis consisted only of refinement of recovery credit.

The analyst estimated the probability for the operator failure to recover the condenser following the MSIV closure as 1.0E-3 using the Accident Sequence Precursor Human Reliability Analysis methodology, which corresponded to an operator recovery credit of 3 when using the Phase 2 worksheets.

After application of the refined operator recovery credit, the dominant accident sequences involved: a TPCS initiating event, failure to recover the main condenser, failure of containment heat removal, and failure to vent the containment; and a TPCS initiating event, failure to recover the main condenser, failure of high pressure injection, and failure to depressurize the reactor. The increase in core damage frequency of the finding due to internal initiating events was greater than 1.0E-8, but less than 1.0E-7. Therefore, the operator error that resulted in the reactor scram with closure of the main steam isolation valves was very low risk (Green). This finding was associated with the cross cutting area of Human Performance, in that a failure to adhere to a procedure resulted in the plant transient.

Enforcement. Technical Specification 5.4, “Procedures,” section 5.4.1 requires in part that written procedures be established, implemented, and maintained covering activities listed in Appendix A to Regulatory Guide 1.33, Revision 2. Contrary to the above, the operators failed to properly implement the written instructions of procedure 2.1.1, “Startup from Shutdown,” by failing to properly operate the pressure regulating system, maintain the required MPR setpoint, and heed the procedure caution. Because the finding was determined to be of very low safety significance and has been entered into your corrective action program (CR 200302159), this violation is being treated as a non-cited violation consistent with section VI.A of the NRC enforcement policy (NUREG 1600). **(NCV 050-293/03-06-01).**

1R15 Operability Evaluations

a. Inspection Scope

The inspector reviewed selected operability determinations to assess the adequacy of the evaluations, the use and control of compensatory measures, compliance with the technical specifications, and the risk significance of the issues. The inspector used the technical specifications, Final Safety Analysis Report, associated Design Basis Documents and PNPS Procedure 1.3.34.5, “Operability Evaluations,” as references. The specific issues reviewed included:

- CR 200301278, EDG Parallel Testing with Single Failure (Standing Order 03-02)
- OE 03-011, RCIC Pressure Switch PS-1360-9C Reset Value (CR 200301358)
- OE 03-023, Drywell Torus Vent System Retains Water (CR 200301417)
- OE 03-024, MSIV AOT-203-1D Failed Operability Test (CR 200302034).

In regard to OE 03-023, the inspector reviewed PNPS engineering evaluation (EE) EE 03-023 which evaluated the impact of standing water in the drywell to torus main vent pipes at the pipe low point or “bowl.” This condition was observed by PNPS during the April 2003 refueling outage and documented in condition report (CR) CR-PNP-2003-01417. The inspector reviewed the description of the PNPS containment and assumed resultant thrust loads during the initial phases of a postulated event to compare the predicted load stresses against the applicable ASME Code allowable stresses and to determine whether the PNPS evaluation was consistent with the approach recommended by the boiling water reactor owner’s group (BWROG). The inspector assessed the PNPS basis for concluding that the torus vent system and containment were operable but degraded. Additionally, the inspector reviewed licensee actions this period to remove the accumulated water in the torus vent system and to evaluate what impact the potential retention of additional water would have on the vent system.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed existing operator workarounds, burdens, and tour checks to assess their cumulative effect on the operators' ability to respond in a correct and timely manner to plant transients and accidents and to implement the abnormal or emergency operating procedures. The inspector also considered the impact of the workarounds on the system(s) reliability, availability, and potential for mis-operation. This inspection was accomplished through discussion with plant operators, plant tours, review of operations performance indicators and supporting data including operability evaluations, temporary modifications, temporary alterations, tagouts greater than 90 days, control room deficiencies and disabled annunciators. Station procedures 1.3.34.4, "Compensatory Measures," and 1.3.34.7, "Operations Performance Indicators," were referenced.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

.1 Annual Inspection

a. Inspection Scope

The inspector selected a risk-significant plant modification package for review to verify that the design bases, licensing bases, and performance capability of the risk significant system had not been degraded through the modification.

For the selected modification, the inspector reviewed the design inputs, assumptions, and calculations to determine the design adequacy. In addition, the inspector reviewed the associated 10 CFR 50.59 safety evaluation to verify that the safety issue pertinent to the changes were properly resolved or adequately addressed. The inspector also reviewed: (1) field revision notices that were issued during the installation to determine proper routing of vent line tubing, stress analyses and tube supports; and, (2) post-modification functional testing to determine the readiness for operations. The inspector reviewed the affected procedures and drawings to verify that the affected documents were appropriately updated. The inspector reviewed licensee actions to disposition discrepancies identified during the modification (reference condition reports

200301670, 1711 and 1745, and Nonconformance Report 03-028). The inspector walked-down portions of the modification in the drywell to review installation conditions. The inspector monitored the performance of the vessel level instrumentations during subsequent periods of plant operations.

The modification package selected for review was:

- PDC 02-114, Reactor Water Level Condensing Chamber Vent Modification

PDC/FRN 01-01-09, “Reactor Recirc Pump Leak Sealant Injection”; was reviewed to ensure that re injection of leak sealant in the A & B recirculation pump jacking bolt holes did not constitute a temporary repair of the primary pressure boundary (which would be prohibited) or effect the structural integrity or functionality of the pumps. The completed maintenance procedures 3.M.4-42, “Furmanite or Equivalent Leak Sealing Process”; were reviewed for each pump to verify the injected quantity of sealant did not exceed the specified maximum allowed.

- Findings

No findings of significance were identified.

.2 Biennial Inspection

a. Inspection Scope

The inspectors reviewed eight selected risk-significant plant modification packages to verify that: (1) the design bases, licensing bases, and performance capability of risk significant Structures, Systems or Components (SSC) had not been degraded through modifications; and, (2) modifications performed during increased risk configurations did not place the plant in an unsafe condition. The modification packages were selected from among the design changes that were completed within the past two years.

The selected plant modifications were distributed among initiating event, mitigating, and barrier integrity cornerstones. For these selected modifications, the inspectors reviewed the design inputs, assumptions, and design calculations, such as instrument set-point, instrument uncertainty, and electrical loading calculations, to determine design adequacy. The inspectors also reviewed field change notices that were issued during the installation to confirm that the problems associated with the installation were adequately resolved. In addition, the inspectors also reviewed the post-modification testing, functional testing, and instrument calibration records to determine readiness for operations. Finally, the inspectors reviewed the affected procedures, drawings, design basis documents, and UFSAR sections to verify that the affected documents were appropriately updated.

The listing of the reviewed modifications is provided in the attachment to this report.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspector reviewed post-maintenance test activities on risk significant systems to verify that the effect of the test on the plant had been evaluated adequately, test equipment was appropriate and controlled, the test was properly performed in accordance with procedures, and the test data met the required acceptance criteria, and the test activity was adequate to verify system operability and functional capability following maintenance. The inspector verified that systems were properly restored following testing and that discrepancies were appropriately documented in the corrective action process. The inspector reviewed the following post maintenance testing (PMT) activities:

- MR 02107411, PMT for MO-1400-24A MCB switch replacement
- MR 03105244, PWT for MO-1001-28A Inspection per 8.Q.3-8.3
- CR 200301337, Secondary Containment Leak Rate Test per 8.7.3 completed after Reactor Building door repair
- Local leak rate testing of feedwater check valves 6-CK-62A and 6-CK-62B completed per 8.7.1.5, "Local Leak Rate Testing of Primary Containment Penetrations and Isolation Valves"
- Post outage testing of the high pressure coolant injection turbine/pump per 8.5.4.3, "High Pressure Coolant Injection Operability Demonstration and Flow Rate Test at 150 psig" and 8.5.4.9, "High Pressure Coolant Injection Turbine Overspeed Trip Test."

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

Review of Outage Plan

The RFO-14 outage risk assessment and procedure TP03-021, “RFO14 Compensatory Measures”; were reviewed to verify that the licensee addressed the outage’s impact on defense-in-depth for the five shutdown critical safety functions; electrical power availability, inventory control, decay heat removal, reactivity control, and containment. Adequate defense-in-depth was verified for each safety function and / or where redundancy was limited or not available, the existence of appropriate planned contingencies, to minimize the overall risk, was verified. The highest noted planned risk condition for the outage was Yellow, reflecting a low to moderate increase in plant risk. The majority of the Yellow condition resulted from conservatism applied to the containment safety function. Consideration of operational experience was also verified. The daily risk up-date, accounting for schedule changes and unplanned activities were also periodically reviewed.

Monitoring of Plant Shutdown and Cooldown Activities

The inspector reviewed licensee action to shut the plant down in accordance with procedures 2.1.14, “Station Power Changes,” and 2.1.5, “Controlled Shutdown from Power.” Portions of various activities to place the plant in a cold shutdown condition and on shutdown cooling were observed by the inspector to assess operator performance, communications, command and control and procedure adherence. Reactor vessel cool down rate, recorded per 2.1.7, “Vessel Heat up and cool down”; was verified within technical specification requirements.

Control Rod Drive Removal and Blade Swap Activities

The inspector reviewed licensee controls to replace control rod drives and blades during the outage. The inspection included a review of licensee procedures 2.2.8.7, “Jumper For Control Rod “Full IN” To Allow Multiple Control Rod Removal during an RFO,” and 3.M.4-1, “Control Rod Drive Removal and Installation,” and tagouts 03-0097 and 03-0098. The inspector reviewed licensee actions to meet Technical Specification 3.10.D (License Amendment 199). The inspector observed activities in progress on April 23, 2003, and noted proper communication and the coordination of activities between the control room, the refueling floor and the under-vessel area during the concurrent fuel shuffle and control rod drive removal activities.

The inspector reviewed licensee actions to replace and swap control rod blades. The control rod blade management strategy assures that control rod worth remains within the design requirements despite depletion and boron loss (reference GE SIL 637, Condition Report 20030736 and Reactor Engineering Memo dated 4/2/3, “RFO 14 Control Rod Blade Replacement / Shuffle”).

Fuel Shuffle Activities and Reactivity Control

The inspector verified that refueling activities were conducted in accordance with the technical specifications and procedure 4.3, "Fuel Handling." Other procedures used during this review are described in the attachment to this report. The inspector observed licensee actions during core alterations to assure core reactivity was controlled. The inspector observed activities from the control room and the refueling floor at various times during the fuel shuffle activities. The inspector verified that the location of fuel and core components was tracked in accordance with the Fuel Movement Schedule and the SHUFFLEWORKS computerized tag board program. The inspector verified licensee action to meet the requirements of Technical Specification 3.10 for core alterations, including the requirements for core monitoring when the A source range monitor became inoperable on April 23, 2003 (reference Condition Report 200301597). The inspector reviewed the licensee's use of and technical bases for alternate core quadrant definitions as described in procedure 4.3. The inspector observed communications and the coordination of activities between the control room, the reactor cavity and the drywell while fuel shuffle and control rod drive removal activities were in progress concurrently.

Tagout Activities

The inspector reviewed tagging activities to verify that the proper tags were hung and that equipment was controlled in accordance with the clearance requirements. The review focused on the tagouts applied to the hydraulic control units during multiple control rod withdrawal for control rod drive and blade replacement. The following tagouts were reviewed:

- 03-0097, East HCUs for Shutdown Margin and TS 4.10.D Requirements
- 03-0098, West HCUs for Shutdown Margin and TS 4.10.D Requirements
- 03-0099, East HCU Accumulator Discharge Boundary
- 03-0100, West HCU Accumulator Discharge Boundary

Other references used for this review are listed in the attachment to this report. The inspectors also reviewed the licensee's actions to strengthen the control of tagging activities (reference Condition Report 200301500) and a fact finding report associated with CR 200301663 pertaining to an error associated with the restoration from tagout 03-0087A which created an unintended drain path from the reactor and spent fuel pool. (See findings below and Section 4OA4)

Reactor Instrumentation

The inspector reviewed reactor temperature, level and pressure indications during the outage to verify conditions were maintained in accordance with license requirements. The inspector reviewed the operator use of a temporary vessel level instrumentation that was installed for the outage, as described in Section 1R23 below (Temporary Alteration 03-01-27).

Shutdown and Spent Fuel Pool Cooling Operations

The inspector reviewed the plant configuration periodically during the outage to verify that spent fuel pool and shutdown cooling system were operated in accordance with licensee procedure 2.2.19.1, Residual Heat Removal System - Shutdown Cooling Mode of Operation. NRC reviews of this area are also described in Section 1R04 above.

Inventory Control

The inspector reviewed licensee actions to establish, monitor and maintain the proper water inventory in the reactor during the outage, and in the reactor and spent fuel pool after flooding the reactor cavity for refueling activities. The inspector reviewed the plant system flow paths and configurations established for reactor makeup and verified the configurations were consistent with the outage plan. The inspector observed operators implement procedure 2.2.19.1 to control reactor inventory during the outage and during the swap of residual heat removal cooling from the B loop to the A loop on April 29, 2003.

Containment Control

The inspector reviewed licensee activities during the outage to control primary and secondary containment. The inspector reviewed testing that demonstrated the secondary containment was operable as required by the technical specifications. The inspector reviewed licensee activities to clean and prepare the containment for closure prior to plant restart.

Plant Startup - Approach to Critical

The inspector observed operator performance during the plant startup activities on May 12, May 18, May 20 and June 2. The inspection consisted of control room observations and a review of the operator logs, plant computer information, station procedures 2.1.1, “Startup from Shutdown,” and 2.1.14, “Station Power Changes.” The inspector observed the approach to critical on May 12 and 18. The inspector verified the licensee action to meet the Technical Specification requirements for compliance with the banked position withdrawal sequence (BPWS) and the rod worth minimizer. The inspector observed the acquisition of data for the in-sequence shutdown margin determination and verified the calculated test results per 9.16.1, “In-Sequence Critical For Shutdown Margin Demonstration,” met the technical specification requirements.

License Amendments and Regulatory Commitments

The inspector reviewed licensee actions to change the conditions of License DPR-35 to support outage maintenance and design change activities. The inspector verified on a sampling basis the licensee’s implementation of technical specification and license amendment requirements during the conduct or outage and startup activities. Licensee activities associated with the following license amendments were reviewed:

- Amendment 201, Thermal Power Optimization
- Amendment 200, Emergency Core Cooling System Requirements
- Amendment 199, Refueling Interlocks and Multiple Control Rod Removal
- Amendment 198, Instrumentation Trip Level Settings and Calibration
- Amendment 197, RCS Pressure-Temperature Curves
- Amendment 196, Relocation of Control Rod Block Function

The inspector reviewed the licensee's plans and schedules to perform inservice inspection activities of reactor vessel welds in accordance with regulatory requirements (reference Entergy letter 2.03.045 dated April 14, 2003, and NRC Relief Request No. 28 dated April 11, 2003). The inspector reviewed the licensee commitments to Generic Letter 88-01 and BWRVIP-75, "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules," and the plans to defer inspection of certain Category D welds until refueling outrage RFO-15 (reference Entergy White Paper on BWRVIP-75 issued April 30, 2003).

b. Findings

Introduction. (Green) The inspectors determined that the unintended drain path from the reactor vessel, which was created due to an inadequate tagout restoration, was a self revealing finding of very low safety significance (Green) and constituted a non-cited violation of technical specification 5.4, "Procedures."

Description. On April 27, 2003, the tagging coordinator modified the restoration sequence for tagout 03-0087A without performing an adequate review of the proposed change or seeking a peer check. As a result of the coordinator's changes, a drain path was created from the reactor vessel via the control rod drive system to the clean radwaste tanks. The gradual drain of the refueling cavity volume went undetected for approximately 7 hours at which point a greater than expected level increase in the clean radwaste tanks was reported. The drain path was identified and terminated approximately three hours later after receiving alarms in the control room due to low level in the fuel pool skimmer surge tank and high level in the scram discharge instrument volume.

Analysis. The finding was greater than a minor issue because it could reasonably be viewed as a precursor to a more significant event and if left uncorrected would become a more significant safety concern. The finding is not greater than Green in accordance with MC 0609 Appendix G, because the level change was less than 2 feet. The actual decrease in refueling cavity level was limited to approximately one inch, despite the 10 hour duration and estimated 3000 gallons of discharged inventory, due to the large available volume of water and operation of the makeup systems. This finding was associated with the cross cutting area of Human Performance, in that operators failed to properly implement the Tagging Procedure.

Enforcement. Technical Specification 5.4, "Procedures," section 5.4.1 requires in part that written procedures be established, implemented, and maintained covering activities listed in Appendix A to Regulatory Guide 1.33, Revision 2. Contrary to the above, the operators failed to properly implement the written instructions of procedure 1.4.5, "PNPS Tagging Procedure," specifically section 6.2.10, by failing to ensure appropriate restoration positions and sequences were specified. Because the finding was determined to be of very low safety significance and has been entered into your corrective action program (CR 200301663), this violation is being treated as a non-cited violation consistent with section VI.A of the NRC enforcement policy (NUREG 1600). **(NCV 050-293/03-06-02)**

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed and observed surveillance testing to verify that the test acceptance criteria was consistent with technical specifications and Updated Final Safety Analysis Report requirements, the test was performed in accordance with the written procedure, the test data was complete and met procedural requirements, and the system was properly returned to service following testing. The inspector observed pre-job briefs for the test activities. The inspector verified that systems were properly restored following testing and that discrepancies were appropriately documented in the corrective action process.

The inspector reviewed the results of the following surveillance tests:

- 8.10.1, Refueling Platform Interlocks Functional Test completed on 4/22/03
- 8.7.3, Secondary Containment Leak Rate Test completed on 4/19/03 (CR 200301337; Tracking LCO 1-03-0125)
- 8.9.8.1, Station Battery Acceptance, Performance and Service test completed on 4/21/03 (CR 200301812, 200301876)
- 8.9.1, A EDG Testing in Parallel Operations (CR200301278)
- 8.7.1.6, local Leak Rate Testing of the Main Steam Isolation Valves on 4/24/03.
- 8.5.4.9, High Pressure Coolant Injection (HPCI) Turbine Overspeed Trip Test
- 8.5.4.3, HPCI Operability Demonstration and Flow Rate test at 150 PSIG.
- 9.16.1, "In-Sequence Critical For Shutdown Margin Demonstration."
- 8.M.3-1, Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loss of Off-Site Power and Special Shutdown Transformer Load Test
- 8.5.1.3, Core Spray Motor-Operated Valve Quarterly Operability Test.

The inspection reviewed the licensee's process to identify and resolve discrepancies identified during the testing program. On May 9, 2003, reactor vessel level was inadvertently lowered approximately 9 inches while performing Procedure 8.M.3-1, "Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loss of Off-Site Power and Special Shutdown Transformer Load Test." The inspector reviewed the test procedure, condition report (CR) 20032010, control room operating logs and discussed the event with operations personnel.

b. Findings

Inadvertent Draindown During ECCS Sequencing / Loss of Off-Site Power Testing

Introduction. A Green self revealing NCV was identified for the failure to have an adequate surveillance procedure in accordance with TS 5.4.1.a which resulted in the inadvertent drain down of the reactor vessel through the automatic depressurization system (ADS) valves to the torus during emergency power source testing.

Description. Procedure 8.M.3-1, “Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loss of Off-Site Power and Special Shutdown Transformer Load Test,” conducts a test on the emergency diesel generators (EDG) and secondary, off-site, AC power source to verify their ability to accept emergency loads within specified time requirements. Procedure 8.M.3-1, Attachment 2, “Special Test Pre-Evolution Brief” section 1[4] states that “This test involves installation of test devices and the defeating of interlocks. There is no vessel refill source available. Control Room personnel shall continuously observe Reactor water level while the CSCS/ECCS injection valves are defeated.” Additionally, procedure 8.M.3-1, “Abort Criteria” section 5[5] states “If any abnormal condition or unexpected response to reactor pressure, power, or level occurs during this test, then terminate the test and investigate the cause. DO NOT proceed with further testing until a cause is known.”

The test was initiated on May 9, 2003, with a reactor water level of 220 inches. This condition corresponds to a vessel level which is above the main steam lines and ADS valves. The procedure initial conditions did not specify a reactor water level band. Attachment 1, section 1[25], note 2, of the procedure, states that “ADS valves will indicate OPEN after the 11-minute ADS timer runs out.” Although this note was discussed during the pre-evolution brief, the operators did not recognize that with the ADS pilot valves energized, and with the reactor vessel water level at 220 inches, there would be sufficient static pressure to cause the ADS valves to unseat. During the test, when the ADS timer timed-out, the reactor water level started to decrease. The operator assigned to monitor the reactor vessel level noted the decrease and informed the Control Room Supervisor (CRS). The verified that the ADS valves indicated open and directed that ADS be inhibited and reset. Reactor vessel level steadied out at 211 inches.

Analysis. The test procedure was not appropriately prepared and resulted in the inadvertent drain down of the reactor vessel from 220 inches to 211 inches. Specifically, Attachment 1, section 1[25], note 2, of the procedure, states that “ADS valves will indicate OPEN after the 11-minute ADS timer runs out.” The procedure does not provide direction to the operators to establish plant conditions to prevent reactor vessel drain down in the event that level is above the ADS valves and the ADS timer times-out. Additionally, the procedure does not provide guidance to the operators to inhibit ADS to prevent the ADS timers from actuating the pilot solenoids and allowing the ADS valves to unseat and drain the vessel.

The drain down through the ADS valves would have self-terminated when the reactor vessel water level reached the main steam lines. The main steam line penetrations in the reactor vessel are above the top of the active fuel. In addition, during the test, a dedicated watch was assigned to continuously monitor reactor vessel water level.

The loss of reactor water level is an issue that is more than minor because it is a precursor to a more significant event, the loss of shutdown cooling. The inspectors applied the guidance of Manual Chapters (MC) 0609, Appendix G, Table 1 for a BWR in cold shutdown or refueling operations with time to boil greater than 2 hours. The finding did not meet the criteria for a finding requiring a phase 2 analysis. Because the loss of level was less than 24 inches, it was not considered a loss of control event. Additionally, the finding did not degrade the licensee's ability to terminate the leak path, recover decay heat removal, if lost, or impact the ability to establish a heat removal path through the SRVs to the suppression pool. Since the finding did not require a quantitative assessment, it is determined to be of very low safety significance (Green) when assessed in accordance with MC 0609, Appendix G.

Enforcement. TS Section 5.4.1 states in part that "Written Procedures shall be established, implemented, and maintained covering the applicable procedures recommended in RG 1.33, Revision 2, Appendix A, February 1978.

RG 1.33, Revision 2, Appendix A, February 1978, item 8b, requires procedures to be maintained for the surveillance tests listed in the TS.

Contrary to the above Procedure 8.M.3-1, Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loss of Off-Site Power and Special Shutdown Transformer Load Test, was not appropriately maintained and resulted in the inadvertent drain down of the reactor vessel from 220 inches to 211 inches. This violation is being treated as a non-cited violation consistent with section VI.A of the NRC enforcement policy. This issue was entered into the corrective action program as CR 20032010. (**NCV 050-293/03-06-03)**.

1R23 Temporary Plant Modifications

Temporary alteration 03-1-006, which installed a temporary 125 Volt battery, charger, and breaker panel; was reviewed to ensure that the temporary system did not adversely effect the system's safety function, was installed consistent with the modification document, and that applicable drawing(s) and/ or operating procedures were up-dated to reflect the temporary alteration. To accomplish the inspection, the inspector walkdown the accessible portions of the modification, reviewed the Updated Final Safety Analysis Report, Technical Specifications, 50.59 screening and referenced safety evaluation S.E. 3364, and temporary procedure TP02-010, "Temporary 125V DC Power Feed For the 125V "A" and "B" DC Systems During Battery Discharge Testing."

The inspector reviewed temporary alteration 03-1-27, "Reactor Vessel Shutdown / Floodup Level for RFO 14." The inspector reviewed the design inputs and calculations to determine the design adequacy. In addition, the inspector reviewed the associated 10 CFR 50.59 safety evaluation screening to verify that the safety questions pertinent to the change were properly addressed. The inspector also reviewed the post-installation testing records to verify that instrument loop accuracies and uncertainties were addressed. The inspector reviewed the use of the temporary level channel during routine control room activities. Finally, the inspector verified the temporary alteration package addressed restoration of the instrumentation channel to its normal configuration at the end of the outage.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control

a. Inspection Scope

During the period from April 28 - May 2, 2003, the inspector reviewed exposure significant work areas, high radiation areas, and airborne radioactivity areas in the reactor (including drywell and refueling floor) and turbine buildings, and evaluated associated controls and surveys of these areas to determine if the controls (i.e., surveys, postings, barricades) were acceptable. For these areas, the inspector reviewed radiological job requirements and attended job briefings to determine if radiological conditions in the work area were adequately communicated to workers through briefings and postings. The inspector also verified radiological controls, radiological job coverage, and contamination controls to ensure the accuracy of surveys and applicable posting and barricade requirements. The inspector determined if prescribed radiation work permits (RWPs), procedure and engineering controls were in place; whether licensee surveys and postings were complete and accurate; and if air samplers were properly located. The inspector conducted reviews of RWPs used to access these and other high radiation areas to identify the acceptability of work control instructions or control barriers specified. The inspector reviewed electronic pocket dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy. Plant technical specification (TS) 5.7 and the requirements contained in 10 CFR 20, Subpart G were utilized as the standard for access control to these areas.

Significant radiological work being performed at the time of this inspection included work activities associated with refueling outage (RFO14) which included: in vessel visual inspection; safety relief valve repair/replacement; in-service inspection; recirculation pump repairs; under vessel work; and, local leak rate testing.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed current ALARA job evaluations, exposure estimates, and exposure mitigation requirements and compared ALARA plans with the results achieved. A review of actual exposure results versus initial exposure estimates for current work was conducted including: comparison of estimated and actual dose rates and person-hours expended; determination of the accuracy of estimations to actual results; and determination of the level of exposure tracking detail, exposure report timeliness and exposure report distribution to support control of collective exposures to determine conformance with the requirements contained in 10 CFR 20.1101(b).

The corporate exposure goal established for RF014 is 150 person-rem, while pre-outage estimation of the work scope was 196 person-rem. Major jobs during RFO14 include: replacement of control rod drives (completed under budget, but with a scope reduction of 10 drives); replace in-board feed water check valves; and, work on the moisture separator reheaters. Through the first two weeks of the outage, total exposures were tracking closely with estimates to reach the 150 person-rem goal.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity including: portable field survey instruments, friskers, portal monitors, and small article monitors. The inspector conducted a review of instruments observed, specifically verification of proper function and certification of appropriate source checks for these instruments, which were utilized to ensure that occupational exposures were maintained in accordance with 10 CFR 20.1201.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

a. Inspection Scope

The initiating event 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 was reviewed to assess the completeness and accuracy of the reported information. Specifically, PI data for the 2nd, 3rd, and 4th quarters of 2002 and 1st quarter 2003 was reviewed and compared to information contained in NRC inspection reports, Licensee Event Reports, and the plant's monthly operating reports. The impact on the PIs for the scrams and unplanned power change which occurred during the 2nd quarter 2003 was reviewed and discussed with station personnel.

b. Findings

No significant findings were identified.

4OA2 Identification and Resolution of Problems

.1 Steam Dryer Leveling Screw Tack Welds

a. Inspection Scope

The inspector reviewed nonconformance report (NCR) 97-124 and indication notification report (INR) 99-06 which identified flaws discovered in the steam dryer leveling screw tack welds during previous outages. The inspector verified that flaws in the steam dryer leveling screw tack welds identified during non-destructive testing were reported, characterized, evaluated and appropriately dispositioned and entered into the corrective action program.

b. Findings

No findings of significance were identified.

.2 Inoperable CHREAFs, Condition Report CR-PNPS-2002-11609

a. Inspection Scope

On September 12, 2002, the humidity switch for the "B" Control Room High Efficiency Air Filtration (CRHEAF) system was wired incorrectly. Instead of turning on a heater whenever the CRHEAF system is running and the humidity is greater than 70%, the incorrect wiring produced the opposite of the intended result and turned the heater off. Entergy did not detect this error in their post maintenance testing and declared the system operable before beginning work on the "A" CRHEAF train. This lead to NCV 50-293/02-07-02, discussed in inspection report 50-293/02-07, section 1R19.

The inspector reviewed Entergy's assessment of the root and contributing causes of this event to ensure that they captured all relevant elements. The inspector also reviewed the corrective actions that were taken to verify that they were appropriately focused and complete. This review included confirmation that the necessary procedure changes were made and that Entergy addressed the human performance and work control aspects of the event as well as the equipment issues.

b. Findings

No findings of significance were identified.

.3 Plant Modifications

a. Inspection Scope

The inspectors reviewed condition reports (CR) associated with 10 CFR 50.59 issues and plant modification issues to ensure that the licensee was identifying, evaluating, and correcting problems associated with these areas and that the corrective actions for the issues were appropriate. The inspectors also reviewed two self-assessments related to 10 CFR 50.59 and plant modification activities at PNPS.

The listing of the condition reports and self assessments reviewed is provided in the attachment to this report.

b. Findings

No findings of significance were identified.

.4 Occupational Radiation Safety

a. Inspection Scope

The inspector reviewed self-assessment reports related to occupational radiation safety, and determined if identified problems were entered into the corrective action system for resolution. Documents reviewed include 37 condition reports of outage related radiation protection program performance, including control of radiologically significant areas, contamination control, and exposure minimization. The inspector also reviewed the tracking, evaluation and resolution of identified issues.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

.1 (Closed) LER 50-293/2001-007: Automatic Scram During Transient Caused by Failure of Calibrating Unit. This item was open pending completion of the long term corrective actions to address level instrument anomalies. Previous NRC review of the events described in the LER were described in Reports 2001-08, 2001-12 an 2002-03. The inspector reviewed the actions completed during this period to modify the vessel level instrument condensing chambers (see Section 1R17 of this report). This LER is closed.

.2 Reactor Scram Events on May 19 and June 1, 2003

The automatic reactor shutdown events of May 19 and June 1, 2003 (see 1R14) were evaluated per Management Directive 8.3, “NRC Incident Investigation Program,” to determine whether the events warranted immediate followup under the event response inspection program, (i.e. Special Inspection, Augmented Inspection Team, or Incident Investigation Team); they did not. The event notifications (EN) required per 10 CFR 50.72 were reviewed and EN 39857 for the May 19th scram was discussed with the plant manager and operations manager who concurred that a revised EN was required to correct information pertaining to the condition of the rods following the scram (CR 200302167). The original report stated five rods had failed to insert on the scram when in fact all rods had fully inserted. The mis-communicated information was that the position indication for five rods did not immediately reflect that the rods had fully inserted as the indicators had gone blank due to the rods going to the over-travel position on the scram. Indication was promptly restored for the five rods in accordance with station procedures and indicated the rods were at the 00 full in position. Immediate corrective actions were instituted by station management to minimize the potential for future mis-communications.

4OA4 Cross - Cutting Issues

Cross-References to Human Performance Findings Documented in the Report

Section 1R14 describes a finding related to the operator failure to follow procedures which resulted in a reactor scram on May 19, 2003 (Condition Report CR CR200302159).

Section 1R20 describes an operator error during a tagging activities on April 27 resulting in an unintended drain of the reactor vessel which continued for approximately 10 hours until it was detected and the drain path identified and secured.

4OA5 Review of Third Party Assessment Reports

The inspector reviewed the results of the Pilgrim Plant Evaluation conducted by the Institute of Nuclear Power Operations (INPO) in November 2002. The inspector noted that the INPO assessment results were consistent with the NRC's assessment of Pilgrim activities.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the plant modification inspection results to Mr. S. Bethay, Engineering Director, and other members of licensee management at the conclusion of the inspection on May 22, 2003. The licensee acknowledged the inspection findings presented.

The inspectors presented the quarterly inspection results to Mr. M. Baldizzi and other members of licensee management at the conclusion of the inspection on July 2, 2003. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered propriety. No propriety information was identified.

4OA7 Licensee Identified Violations

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a non-cited violation.

- Appendix B, Criterion IX, "Control of Special Processes," to 10 CFR 50, requires that special processes, including welding are controlled and accomplished using qualified procedures in accordance with applicable codes and standards. Contrary to the above, the licensee, implemented welding procedure specification, P1-TS, without ensuring that the welding procedure specification was properly qualified by supporting procedural qualification records as required by ASME Section IX. This was identified in the licensee's corrective action program as condition report (CR) 200301071. This finding was of very low safety significance because the licensee was able to subsequently demonstrate that the welding procedure specifications were acceptable.

- 10 CFR 50.9, "Completeness and Accuracy of Information," requires, in part, that information provided to the commission, by a licensee, shall be complete and accurate in all material respects. Contrary to this, on April 10, 2003, Entergy notified the NRC that one reactor operator license renewal had been certified by Entergy with inaccurate information and renewed by the NRC based on the inaccurate information (CR-PNPS-2003-01269). The finding is of very low safety significance because, even though the individual was not current in requalification training, he had completed the comprehensive written examination and annual operating examination prior to renewal of his license, therefore demonstrating his ability to properly discharging his licensed responsibilities.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

S. Bethay	Director of Engineering
S. Brennion	Licensing Superintendent
S. Das	Senior Lead Engineer, Design Control
G. Dyckman	Senior Engineer
L. Foreaker	ALARA Specialist
P. Harizi	Senior Lead Engineer
R. Levin	Senior Engineer, Design Control
W. Lobo	Nuclear Safety / Licensing Specialist
J. MacDonald	Control Room Supervisor
W. Mauro	ALARA Supervisor
G. Mileris	Senior Lead Engineer, Design Control
F. McGinnis	Senior Engineer
F. Mogolesko	Power Up-rate Project Manager
K. Mulligan	Engineering Manager
E. Olson	Operations Manager
D. Perry	Radiation Protection Manager
E. Sanchez	Senior Engineer, Design Control
M. Santiago,	Superintendent Operations Training

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

50-293/03-06-01	NCV	Operator Error Caused MSIV Closure and Reactor Scram
50-293/03-06-02	NCV	Inadequate Tagout Resulted in Vessel Drain Event
50-293/03-06-03	NCV	Inadequate Procedures Used to Conduct Emergency Power Source Testing

LIST OF DOCUMENTS REVIEWED

References for Section 1R02

10 CFR 50.59 Safety Evaluations

SE 3295	Revision to Procedure No. 2.4.143 “Shutdown from Outside Control Room”
SE 3329	Change EDG Fuel Consumption Rate
SE 3379	RP-01-045, Temporary Battery Charger
SE 3385	EDG Timer 162A-509(609)
SE 3388	Installation of Relays SE1 and SE2 in Y10 Automatic Transfer Switch
SE 3394	RP-01059 Special Test for the EDG Governor Adjustment or Replacement Postwar Testing
SE 3395	PDC 01-20, Main Steam Line Safety Relief Valves (SRVs) Modification

10 CFR 50.59 Screening Evaluations

FRN01-02, Calculation PS140
FRN02-43-01, Provide Core Bores and 3 hr. Fire Wrap for A2620 in Control Room
FRN02-53-01, TE/TT34003A, B, C, D Calibration Range Change
FRN02-53-03, Power Up-Rate EPIC Software Changes and AMAG System Setup
FRN02-53-04, RTD Bench Testing Change
FRN02-53-06, Power Up-Rate EPIC Software Changes
IN1-74, Rev2, Setpoint Calculation for EDG Day Tank Level Controls
8.M.2-1.5.2, Main Steam Isolation Valve Logic - Test B - Outboard
PDC01-015, Procedures 2.2.80, 2.2.87 and 3.m.2-12.6
PDC01-21, Main Steam and SRV Discharge Line Pipe Support Modification
PDC02-43, EDG KWS Upgrade
PDC 02-53 AMAG/EPIC Interface Modification,
PDC02-139, Change % Regulation and Nominal Voltage for Regulating Transformers
TP01-055, Residual Heat Removal, Shutdown Cooling Mode
Microloc 08815-0395, Reactor Vessel Top Head Flange Leak Detection

References for Section 1R04
M243, High Pressure Coolant Injection System, Sheet 1
M244, High Pressure Coolant Injection System, Sheet 2
M245, Reactor Core Isolation Cooling System, Sheet 1
M245, Reactor Core Isolation Cooling System, Sheet 2
M231, Fuel Pool Cooling & Demineralizer system,
M241, Residual Heat Removal System, Sheet 1&2
2.2.19.1, “Residual Heat Removal System - Shutdown Cooling Mode of Operation”
2.2.85, “Augmented Fuel Pool Cooling (With Shutdown Cooling) Mode 1"
2.2.19,” Residual Heat Removal”
8.C.13,”Locked Component Lineup[Surveillance”
PNPS System Report Card - RHR 2002 4th quarter.

References for Section 1R06
UFSAR Section 10.7.6, “Salt Service Water System Safety Evaluation”
Safety Evaluation Report (SER) 50-84, Internal Flooding Analysis”
Annunciator Response Procedure (ARP) Panel 904L, F6, “RBCCW Pump Area Leakage”
Drawing M59, “Torus and Reactor Aux Bay Area Dewatering Lines”

Reference for Section 1R08
NDT Examination Reports
EDS 03-E-125, Pipe to Pipe, 10-0-24, UT
EDS 03-C-125, Pipe to Pipe, 10-0-25, UT
EDS 03-E-069, Pipe to Nozzle, HL-23-F22, UT
03-P-005, Pipe to Pipe, Shop Welds 2, 3, 4, 5 and 6, PT
03-P-126, Pipe to Valve, DC-10-F9, PT
03-E-128, Pipe to Valve, DC-10-F9, UT
03-M-078, Pipe to Nozzle, HL-23-F22, MT
03-0079, Pipe to Pipe, FW 02119147-1, PT
03-U-052, UT Thickness Examination of Spool JF 29-2-1,2-2 and 2-4A (SSW)

Calibration Data Sheets
CDS 03-C-124, Austenitic Stainless Steel to 1.032T-20"
CDS 03-C-127, Austenitic Stainless Steel to 1.156T-18"
CDS 03-C-068, Ferritic Carbon Steel to 0.375-16"

NDT Examination Procedures
50.87, R1, Ultrasonic Examination of Class 1, 2 and 3 Austenitic Piping Welds

In Vessel Remote Visual Examination
VT-3, Steam Dryer Structural Welds and Leveling Screws

Repair-Replacement
MR 01110262, Shutdown Cooling Suction Valve MO-1001-50 Bonnet Drain Valve
MR 03104029, Repair of Canopy Seal Weld on Recirculation Pump "B"

Flaw Evaluation
UT Report 621030, UT Examination Summary Sheet, N2B Nozzle to Vessel Weld, RPV Recirculation Inlet

Welding Procedure Specification
P1-TS, Gas Tungsten/Shielded Metal Arc of P1 to P1

Nonconformance Reports
NCR 97-124, Defects in Leveling Screws on Steam Dryer

Condition Reports
CR-2003-01618, Indications of Containment Coating Failure
CR-2003-01071, Absence of Post Weld Heat Treatment Qualification for WPS P1-TS
CR-2003-01068, Discrepancy of Coupon Thickness for WPS P8-P1-AT-AG and P8-AT-AG
CR-2003-01472, Steam Dryer Leveling Screws Indications

References for Section 1R11
Emergency operating procedure EOP-01, RPV Control
Emergency operating procedure EOP-02, RPV Control, Failure to Scram
Operator Training Plan O-RQ-06-02-92, EOP-02 Failure to Scram
Procedure 5.3.23, Alternate Rod Insertion
Procedure 5.3.35, Operations Management Emergency and Transient Response Expectations for Operating Crews

References for Section 1R15
M245, Reactor Core Isolation Cooling System, Sheet 1
M245, Reactor Core Isolation Cooling System, Sheet 2
M1G 12-12, RCIC System Elementary Diagram (RCIC Steam Line Low Pressure Isolation)
M1G 15-9, RCIC System Elementary Diagram (Valve 1301-16 and 1301-17)
M1G 2-5, RCIC Functional Control Diagram
Technical Specification 3.5.D and Table 3.2.D

UFSAR 7.3.4.7.9, RCIC Steam Line Low Pressure

References for Section 1R17

Modifications

PDC 01-09	SSW Discharge Piping Cured-in-Place Pipe (CIPP)
PDC 01-21	Main Steam and SRV Discharge Line Pipe Support Modification
PDC 01-20	Main Steam Line Safety Relief Valves (SRVs) Modification
PDC02-43	EDG KWS Upgrade
PDC 02-53	AMAG/EPIC Interface
PDC 02-139	Change % Regulation and Nominal Voltage for Regulating Transformers
PDC 02-165	Replacement of Feedwater Level Control Modules

Procedure

2.4.143 Shutdown from Outside Control Room

Self-Assessments

Self Assessment of the PNPS 50.59 Process dated October 10, 2002
Effectiveness of the Minor Modification Group, dated June 26, 2002

Corrective Action Reports

CR-PNPS-2001-02301, CR-PNPS-2001-03423, CR-PNPS-2001-04409, CR-PNPS -2002-09212, CR-PNPS-2002-09929, CR-PNPS-2002-10159, CR-PNPS-10212, CR-PNPS-2002-11342, CR-PNPS-2002-11391, CR-PNPS-12134, CR-PNPS-2002-12284, CR-PNPS-2002-10874, CR-PNPS-2003-00327, CR-PNPS-2003-00716, CR-PNPS-2003-02010

Procedures

ENN-LI-100	Process Applicability Determination, Revision 3
ENN-LI-101	10CFR50.59 Review Process, Revision 3
1.3.34	Conduct of Operations, Revision 82
2.4.143	Shutdown from Outside the Control Room, Revision 27
8.M.3-1	Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loss of Off-Site Power and Special Shutdown Transformer Load Test, Revision 33
8.5.6.2	Special Test for ADS System Manual Opening of Relief Valves, Revision 18

Miscellaneous

BECo Ltr. No. 96-040	Proposed Technical Specification Changes, dated April 25, 1996
BECo Ltr. No. 2.96-080	Response To Request For Additional Information Regarding Diesel Generator Allowed Outage Time Technical Specification Change (TAC No. M95277), dated September 5, 1996
BECo Ltr. No. 2.97-082	Supplement to Emergency Diesel Generator Allowed Outage Time Technical Specification Change, dated August 8, 1997
BECo Ltr. No. 2.98.030	Supplement to Emergency Diesel Generator Allowed Outage Time Technical Specification Change, dated March 26, 1998
Calc M-1031, Rev. 1	SSW Discharge Piping CIPP Liner Design

ENGC Ltr. 2.01.095	Report of Changes, Tests and Experiments Performed at Pilgrim Nuclear Power Station
Eval 860	PNPS Commercial Grade Item Engineering Evaluation for CIPP Lining

References for Section 1R20

3.M.1-45, Outage Shutdown Risk Assessment
1.5.22, Risk Assessment Process
TP03-021, RFO14 Compensatory Measures
QA Surveillance Report 02-068, RFO-14 Initial Outage Shutdown Risk Assessment
RFO-14 Final Risk Assessment Report
Power Maneuvering Plan PMP-MAN.C14-25 dated 4/14/3
OPER-13, Daily Refueling Checklist
OPER-14, Shift Refueling Checklist
OPER-25, Fuel Movement Within the Spent Fuel Pool Checklist
2.2.19.1, Residual Heat Removal System - Shutdown Cooling Mode of Operation (Rev.7)
2.1.7, Vessel Heatup and Cooldown (Rev 43)
2.4.25, Loss of Shutdown Cooling (Rev 23)
2.2.85.1, Augmented Fuel Pool Cooling (With Shutdown Cooling) Mode 1 (Rev 6)
Technical Specification 3.10.A and 4.10 A, Refueling Interlocks
Technical Specification 3.10.B, Core Monitoring
UFSAR Section 7.5.4 and Appendix G
ENN-LI-101, 50.59 Screening for Procedure Change to 4.3 Fuel Handling to Incorporate Revised SRM Quadrant Definitions including Rotation of Quadrants
GE-NE-0000-0014-5292, Pilgrim SRM Quadrant Definition Analysis
Material Balance and Accountability Transfer Form: 1862 FILE: RFO14S01
Condition Report 200301500, HCU tagout for refueling maintenance offload
Condition Report 200301812, A 125vdc battery service test results
Condition Report 200301597, SRM ‘A’ Spiking During Core Offload
Calculation C15.0.3381, Allowable Additional Secondary Containment Leakage Area and Gaps at Secondary containment Doors, in support of procedure 8.7.3
Engineering Memorandum EED03-002 for A 125 vdc Battery Testing dated 2/12/03
3.M.4-1, Control Rod Drive Removal and Installation
2.2.87.4, Jumper for Control Rod “Full In” To Allow Multiple Control Rod Removal During RFO
1.3.34.10, Shift Operations Management (SOMs) Clearance Module
1.3.34.12, Shift Operations Management (SOMs) Configuration Control Module

LIST OF ACRONYMS

ALARA	As Low As is Reasonably Achievable
AMAG	Advanced Measurement and Analysis Group
ASME	American Society of Mechanical Engineers
BVOJ	bypass valve opening jack
CFR	Code of Federal Regulations
CIPP	Cured-in-Place Pipe
CR	Condition Report
CRD	Control Rod Drive
CRS	control room supervisor
CS	Core Spray
ECCS	Emergency Core Cooling System
EDG	emergency diesel generator
EE	Engineering Evaluation
EOP	Emergency Operating Procedure
EPIC	emergency and plant information computer
FRN	field revision notice
FW	Field Welds
HPCI	High Pressure Core Injection
INPO	Institute of Nuclear Power Operations
ISI	Inservice Inspection
MCB	Main Control Board
MPR	Mechanical Pressure Regulator
MR	Maintenance Request
NCR	Nonconformance Report
NCV	Non-Cited Violation
NDE	Nondestructive Examination
PASS	Post Accident Sampling System
PDC	permanent design change
PMT	Post Maintenance Test
PNPS	Pilgrim Nuclear Power Station
PT	Penetrant Test
RBCCW	Reactor Building Closed Cooling Water
RG	Regulatory Guide
RHR	Residual Heat Removal
RWP	Radiation Work Permit
SE	safety evaluation
SDP	Significance Determination Process
SRV	safety relief valve
SSW	salt service +water
RP	test procedure
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing