January 29, 2004

EA No. 04-021

Mr. Michael Balduzzi
Site Vice President
Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: PILGRIM NUCLEAR POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000293/2003011 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Balduzzi:

On December 31, 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 January 8, 2004, with Mr. Pete Dietrich and members of your staff.

The inspection 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 two findings of very low safety significance (Green), which involved violations 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 (NCV), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, one licensee-identified violation which was determined to be of very low safety significance is listed in Section 4OA7 of this report. 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 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, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Pilgrim.

In addition, the inspectors reviewed the events relating to the discovery of reactor coolant system pressure boundary leakage (self-revealing) in reactor vessel nozzle N10. Although this issue constitutes a violation of NRC requirements, we have concluded that the degraded condition was not avoidable by reasonable quality assurance measures or management controls and, thus, no performance deficiency was identified. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy and refrain from issuing enforcement action for the violation. An evaluation was performed and we have determined that this was an issue of very low safety significance. The NRC plans to issue an Information Notice discussing this event.

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 were completed 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, 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/

A. Randolph Blough, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000293/2003011
w/Attachment: Supplemental Information

cc w/encl: G. Taylor, Chief Executive Officer
M. Kansler, President, Entergy Nuclear Operations, Inc.
J. Herron, Senior Vice President and Chief Operating Officer
S. Bethay, Director, Nuclear Assessment Group
D. Pace, Vice President, Engineering
B. O'Grady, Vice President, Operations Support
J. McCann, Director, Nuclear Safety Assurance
C. Faison, Manager, Licensing
Director of Oversight, Entergy Nuclear Operations, Inc.
D. Tarantino, Nuclear Information Manager
B. S. Ford, Manager, Licensing
J. Fulton, Assistant General Counsel, Entergy
S. Lousteau, Treasury Department
R. Walker, Department of Public Health, Commonwealth of Massachusetts
The Honorable Therese Murray
The Honorable Vincent deMacedo
Chairman, Plymouth Board of Selectmen
Chairman, Duxbury Board of Selectmen
Chairman, Nuclear Matters Committee
Plymouth Civil Defense Director
D. O'Connor, Massachusetts Secretary of Energy Resources
J. Miller, Senior Issues Manager
Office of the Commissioner, Massachusetts Department of Environmental Protection
Office of the Attorney General, Commonwealth of Massachusetts
Chairman, Citizens Urging Responsible Energy
S. McGrail, Director, Commonwealth of Massachusetts, SLO Designee
Electric Power Division
R. Shadis, New England Coalition Staff
Commonwealth of Massachusetts, Secretary of Public Safety

Distribution w/encl: H. Miller, RA/J. Wiggins, DRA
J. Jolicoeur, RI EDO Coordinator
L. Dudes, NRR
C. Anderson, DRP
D. Holody, ORA
R. Urban, ORA
F. Arner, DRP
J. Clifford, NRR
T. Tate, PM, NRR
R. Ennis, Backup PM, NRR
W. Raymond, SRI - Pilgrim
C. Welch, RI - Pilgrim
A. Ford, Site Secretary - Pilgrim
Region I Docket Room (with concurrences)

DOCUMENT NAME: G:\BRANCH5\2-PG\PIL0311wjr.wpd
After declaring this document "An Official Agency Record" it **will** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP		RI /DRS (EA	N	OEEA		RI/DRP	
NAME	Wraymond/CJA for		CAnderson/CJA		RLorsen/RL		Dholody/RJU for		ABlough/AB	
DATE	01/28/04		01/28/04		01/28/04		01/28/04		01/29/04	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	05000293/2003011
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	September 28 - December 31, 2003
Inspectors:	W. Raymond, Senior Resident Inspector C. Welch, Resident Inspector J. Furia, Senior Health Physicist Dr. J. Jang, Senior Health Physicist J. H. Williams, Sr. Operations Engineer E. Harold Gray, Sr. Reactor Inspector T. Burns, Reactor Inspector
Approved By:	Clifford Anderson, Chief Projects Branch 5 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000293/2003011; 09/27 - 12/31/03; Pilgrim Nuclear Power Station; Personnel Performance During Non-routine Plant Evolutions; Refueling and Other Outage Activities; Event Follow-up.

The report covered a 14-week period of inspection by resident and region-based inspectors. 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. Inspector Identified and Self-Revealing Findings

Cornerstone: Initiating Events:

- Green. A human performance error resulted in the inadvertent trip of the B reactor recirculation pump and subsequent plant transient and constituted a non-cited violation of Technical Specification 5.4, "Procedures." The failure to implement a maintenance procedure is an example of a contributing cause in the cross cutting area of human performance.

  This self-revealing finding is greater than minor because it had an actual impact on plant stability by causing an unanticipated power reduction. The finding is not greater than very low safety significance (Green) because it did not increase the likelihood of a fire, flood, or LOCA; nor did it result in both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would not be available. (Section 1R14)

Cornerstone : Mitigating Systems:

- Green. The failure to provide adequate guidance in HPCI maintenance procedures resulted in the HPCI system being inoperable for repairs for about 3 days during plant operations. This issue constitutes a self-revealing finding of very low safety significance (Green) and a non-cited violation of Technical Specification (TS) 5.4, "Procedures." The failure to adequately incorporate operating experience in maintenance procedures is an example of a contributing cause in the cross-cutting area of problem resolution. (Section 4OA3)

B. Licensee Identified Violations

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

REPORT DETAILS

Summary of Plant Status

The plant began the period with a power reduction to perform a condenser thermal backwash, followed by a planned outage to install the repaired unit auxiliary transformer, replace control rod drive 30-35, and to investigate and repair sources contributing to the drywell leakage. Following thermal backwash, the turbine was taken off line at 11:00 p.m. on September 29, all control rods were inserted at 4:18 a.m. on September 30, and the mode switch was in refuel at 4:27 a.m. The plant entered cold shutdown on RHR at 6:21 p.m. on September 30. While investigating drywell leakage sources on October 1, the licensee identified reactor coolant system pressure boundary leakage as a through-wall leak in the nozzle to end-cap weld on vessel nozzle N10. The licensee completed an overlay weld repair of the N10 nozzle. The reactor was critical at 9:00 a.m. on October 10, and RCS temperature was heated up above 212 degrees F at 10:11 a.m. The turbine-generator was synchronized with the grid at 12:51 a.m. and the UAT was placed in service at 2:12 a.m. on October 11. The unit returned to routine operations at full power at 4:32 a.m. on October 12.

Plant power was reduced to 65% at 7:01 p.m. on December 1 after the trip of the B recirculation pump when technicians secured from LPCI logic test. The B recirculation pump was restarted at 11:22 p.m. and the unit returned to 100% power at 1:36 p.m. on December 2. Plant power was reduced to 77% at 9:10 p.m. on December 6 when the offsite dispatcher ordered PNPS to reduce load due to an electrical grid emergency in progress. The dispatcher lifted restrictions at 11:30 p.m. and power was restored to 100% at 6:03 a.m. on December 7.

Except as noted above, Pilgrim operated during the period at 100 percent (%) core thermal power for the remainder of the period, except for short periods of planned operation at reduced power for routine testing and maintenance.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope (1 sample)

The inspector reviewed licensee preparations and actions per procedure 2.1.37, “Coastal Storm - Preparations and Actions," when winter storm Alex impacted the east coast during the period of December 5 through 8, 2003. The inspector discussed the implementation of the procedure with Operations personnel, along with the actions taken in response to degraded grid and power supply conditions (reference Condition Reports 200304450 and 200304451).

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

1. Partial System Alignment

a. Inspection Scope (3 samples)

The inspector completed a partial review of risk significant plant systems during periods when the redundant system was out of service for scheduled preventive maintenance and testing. The inspector reviewed plant procedures, system drawings and valve line-up procedures to walkdown and verify the correct system lineup. The Updated Final Safety Analysis Report and the Technical Specifications were reviewed to ascertain the required system configuration. The references used for this review are described in the attachment to this report. This inspection covered 3 samples.

- A EDG System walkdown during B EDG Testing on October 22, 2003
- Electrical System Alignment for Startup Transformer Outage November 3, 2003
- CRD System Walkdown during Rod 30-35 Repair - October 1, 2003

The inspector completed a partial review of the control rod drive (CRD) hydraulic system during periods when control rod 30-35 was declared inoperable and isolated for maintenance on the control rod drive. The inspector conducted a walkdown of the control rod drive hydraulic system to verify that rod 30-35 and the associated 5 X 5 array around the rod were properly isolated in accordance with tagout 03-0054. The inspector reviewed the CRD system drawing M250 to walkdown and verify the correct lineup. The Updated Final Safety Analysis Report and the Technical Specifications were reviewed to ascertain the required system configuration.

b. Findings

No findings of significance were identified.

2. Complete System Walkdown

a. Inspection Scope (1 sample)

The inspector conducted a complete system walkdown of the safety-related portions of the high pressure coolant injection (HPCI) system. The inspection included reviews of the system normal operating procedures 2.2.21 and 2.2.21.5, Drawings 243 and 244, Updated Final Safety Analysis Report Section 6.5.2.3, and the plant technical specifications. The inspector performed a system line-up review including verifying system valves and electrical breakers were in the proper line-up condition. The inspector interviewed licensee personnel and reviewed the status of open work orders, problem reports, temporary modifications, the system health report, and operability evaluations to assess any outstanding deficiencies in the HPCI system. Other references used for this review are included in the attachment to this report.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope (7 Samples)

The inspector toured selective 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. Selected documents reviewed are listed in the attachment. The inspection covered 7 samples.

- Fire Zone 3.5, Vital motor generator set room
- Fire Zone 1.13, Fuel pool pumps and heat exchanger area
- Fire Zone 1.14, Open area and Fan Rooms 74' Reactor Building
- Fire Zone 3.2, Cable Spreading Room (PR 0109665)
- Fire Zone 1.3, HPCI Pump/Turbine Room
- Fire Zone 1.4, HPCI Control Panel Room
- Fire Zone 1.3A, HPCI Blowout Panels Area

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope (1 sample)

The following inspection activities were performed using NUREG-1021, Rev. 8, “Operator Licensing Examination Standards for Power Reactors,” Inspection Procedure Attachment 71111.11, “Licensed Operator Requalification Program,” and NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP),” and 10 CFR55.46, ”Simulator Rule,” as acceptance criteria.

The inspectors reviewed documentation of operating history since the last requalification program inspection. Documents reviewed included NRC Plant Issue Matrix and licensee condition reports. The following condition reports were reviewed for possible training deficiencies and corrective actions:

CR-PNP-200209568
CR-PNP-200210512
CR-PNP-200210627
CR-PNP-200300513
CR-PNP-200303026
CR-PNP-200303218
CR-PNP-200303225

The inspectors did not identify any operational events that were indicative of possible training deficiencies.

The comprehensive written exams and operating tests given in 2002 were reviewed. The inspectors observed the administration of the annual operating test for one operating crew during the week of September 8, 2003. The quality of the written exams, the annual operating tests and the administration and evaluation of the operating tests met or exceeded the criteria of the Examination Standards and 10 CFR 55.59.

The inspectors observed simulator performance during the conduct of the examinations, reviewed simulator performance tests and discrepancy reports to verify compliance with the requirements of 10CFR55.46.

The inspectors reviewed a sample of operators' records related to requalification training attendance, remediation of failures, exam performance, license reactivations, and medical examinations and confirmed the operators were in compliance with license conditions and NRC regulations.

Licensed operators were interviewed for feedback regarding the implementation of the licensed operator requalification training program.

On November 7, 2003, the inspectors conducted an in-office review of licensee annual operating tests results for 2003. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)." The inspectors reviewed the following:

- Operations Crew failure rate;
- Individual failure rate on the dynamic simulator test;
- Individual failure rate on the walk-through test;
- Individual failure rate on the comprehensive biennial written exam;
- Overall failure rate among individuals for all portions of the exam

b. Findings

No significant findings were identified.

1R12 Maintenance Rule (71111.12)

a. Inspection Scope (2 samples)

The emergency diesel generators are within the scope of the maintenance rule 10 CFR 50.65 program. The inspector reviewed problems involving the emergency diesel generators identified in the licensee problem identification and resolution program to assess the performance history of the diesels for the period from November 2002 to December 2003. The review included a consideration of the maintenance rule basis and guideline documents, system health reports, and the updated final safety analysis report (UFSAR). The inspector assessed the effectiveness of licensee actions to monitor the system in accordance with the requirements of 10 CFR 50.65(a)(1) and (a)(2). The inspector focused on the proper classification of selected equipment issues and potential common mode failures, including the following potentially significant Condition Report issues: 200212613 (incorrect start air solenoid); 200300550 (high crankcase exhaust); 200301030 (control cabinet ventilation); 200301278 (operation in parallel with the offsite grid); 200301405 (high air manifold pressure); 200302956 (slow start time); 200303483 (droop relay failure); and, 200304119 (low turbo air assist tank pressure). The inspector verified that the corrective actions were reasonable and appropriate. The inspector confirmed the emergency diesel generators were properly classified as maintenance rule category a(2). This review covered one sample.

Maintenance effectiveness for the DC electrical distribution system was also reviewed for the period 2000 - 2003. This inspection focused on the 250 Volt portion of the DC system. The review included consideration of the maintenance rule basis and guideline documents, system health reports, and the updated final safety analysis report (UFSAR). Condition reports and maintenance records were reviewed for the three year period. System performance in comparison to established availability and functional failure goals was assessed and proper monitoring of the system in accordance with the requirements of 10 CFR 50.65(a)(1) and (a)(2) was verified. The inspector reviewed the DC electrical distribution system to determine if it was properly classified as maintenance rule category a(1). This review covered one inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (3 samples)

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 maintenance activities:

- MR03110875, CRD 30-35 Replacement per TP03-036 and TP03-037
- Surveillance procedure 3.M.3-47 Load Shed Relay Operational/Functional Test (Attachment 1) Friday October 17, 2003
- MR03112659, Startup Transformer Outage for Sprinklers and Other Work on November 2 and 3, 2003

This review covered 3 samples. The references used during this review are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14)

1. Plant Startup and Approach to Critical

a. Inspection Scope (1 sample)

The inspector assessed the control room operator performance during the planned, non-routine evolution: plant startup and approach to critical on October 10 following the outage to replace the unit auxiliary transformer and address drywell leakage. This review covered one sample.

The inspector evaluated personnel performance during this evolution (i.e., adequacy of operator actions, procedure compliance) against the requirements and expectations contained in station procedure 2.1.4, "Approach to Critical." The inspection consisted of control room observations and a review of the operator logs and plant computer information. 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.

b. Findings

No findings of significance were identified.

2. Inadvertent Trip of the B Reactor Recirculation Pump

a. Inspection Scope (1 sample)

On December 1, 2003, while operating at 100% power, a human performance error resulted in the inadvertent trip of the B reactor recirculation pump and subsequent plant transient. The inspector reviewed the events which led to the pump trip and the operator actions taken in response to assess personnel performance, procedure adherence, and equipment response. The review was accomplished via discussion with plant personnel, and review of operator logs, station procedures, and plant computer information.

b. Findings:

Introduction

Green. The inadvertent trip of the B reactor recirculation pump and subsequent plant transient constituted a self-revealing finding of very low safety significance (Green) and a non-cited violation of Technical Specification (TS) 5.4.1, "Procedures."

Description

On December 1, 2003, a fire at the canal electric plant resulted in the loss of the offsite 23 KV line to the Pilgrim shutdown transformer and a blackout on Cape Cod. In response to the instabilities on the electric grid, the shift manager conservatively directed I&C personnel to back out of surveillance procedure 8.M.2-2.10.2-17, " LPCI Break Detection Logic Functional Test, Injection Valves Interlock Test Division B," and restore the safety system to its standby status.

I&C technicians failed to implement the requirements contained in procedure 1.5.17, "Conduct of Maintenance"; for backing out of a procedure. Specifically, contrary to the requirements of procedure 1.5.17, section 6.2.1, Guidelines for backing out of a Procedure, a written or verbal plan was not developed and independently reviewed, appropriate schematics and logic diagrams were not reviewed, and the need to reset relays was not appropriately assessed. As a result, seal-in relays in the LPCI loop select logic, which had picked up during the test were not identified and properly reset prior to restoring the test switches to normal. The result was the automatic closure of the B recirculation loop discharge valve and a trip of the B recirculation pump and the accompanying plant transient.

Analysis

The issue is more than minor because it had an actual impact on stable plant operations. Using MC-0609, the finding screened to Green in Phase 1 of the SDP under the Initiating Event Cornerstone. Specifically, the finding did not increase the likelihood of a fire, flood, or LOCA nor did it result in both the likelihood of a reactor trip and the likelihood that mitigating equipment or functions would not be available.

The failure to follow procedures is an example of a contributing cause in the cross cutting area of human performance, which is discussed in Section 4AO4 of this report.

Enforcement

Technical Specification (TS) requirement 5.4.1 states, in part, that written procedures shall be established, implemented, and maintained covering the activities of Regulatory Guide (RG) 1.33. Contrary to TS 5.4.1, I&C technicians failed to implement procedure 1.5.17, “Conduct of Maintenance,” when they backed out of the surveillance test. Because the finding is of very low safety significance (Green) and has been entered into the corrective action program (CR 03-04387), this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy (NUREG 1600). **NCV 050-293/03-11-01.**

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (7 Samples)

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 Procedures 1.3.34.5, “Operability Evaluations,” and ENN-OP-104, “Operability Determinations,” as references. This review covered 7 inspection samples.

- CR 200303953, X58 Regulating Transformer Output Voltage High
- CR 200303302, HPCI Operability - August 29 Test Failure
- OE 03-027, HPCI Ramp Signal Generator (CR 200303321, 200303305)
- CR 200304081, Control Rod 30-35 Exceeded the Nuclear Design Limits
- CR 200304555, D230 Control Rod Boron Depletion
- CR 200303932, MSIV AO-203-2A packing leakage
- CR 200304197, Reactor Building inner truck lock door gaps

For Condition Report 200304081, the inspector reviewed the licensee’s technical bases for the conclusion that control rod 30-35 was degraded but operable for its reactivity worth function. Control rod 30-35 was declared inoperable due to a malfunction of its control rod drive mechanism, and the rod is electrically disarmed at position 00 per Technical Specification LCO 3.3.B.1.C. The licensee projected that control blade 30-35 will exceed its nuclear and mechanical design life limits as it remains inserted for the remainder of the Cycle 15 operating cycle. The inspector verified the licensee evaluated the effect of the depleted blade on the safety function for the design shutdown margin, compliance with Technical Specification 3.3.A.1/SR4.3.A.1 limits for shutdown margin, core thermal limits, cycle 15 energy, maximum subcritical banked withdrawal position and the banked position withdrawal sequence, the control rod drop analyses, the transient and accident analyses, control blade structural integrity, high control rod notch worth on startup, and reactor water chemistry. The inspector confirmed the licensee evaluation of shutdown margin for rod 30-35 included consideration of the boron depletion in D230 control blades as addressed in GE SIL 637R1 (CR 200304555). The inspector verified that the licensee demonstrated that the impact of the degraded condition in the above areas would be acceptable. Other references used in this review are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope (4 Samples)

This review covered three inspection samples of a specific operator workarounds. The inspector reviewed licensee actions to address items #248, 251 and 253 in the list of operator compensatory measures. The inspector reviewed the deficiencies to determine if the functional capability of the system or human reliability in responding to an initiating event was affected. The inspector evaluated the effect of the deficiency on the operator's ability to implement abnormal and emergency operating procedures.

This inspection covered one inspection sample of the cumulative effects of operator workarounds. The inspector reviewed the operator work around, burden, and tour lists to evaluate the potential cumulative impact of the equipment deficiencies on the operators' ability to implement abnormal or emergency operating procedures. The inspector walked down the control room panels and selected plant areas to review the impact of the deficiencies and to ensure that applicable deficiencies were captured in the licensee's deficiency list. The inspector discussed the operator workarounds with licensee personnel to assess the aggregate impact on plant operations. During the review, the inspector used the criteria contained in licensee procedure 1.3.34.4.

The inspector verified that the licensee evaluated deficiencies for potential impact as operator workarounds and was entering them into the corrective action process. The inspector noted the licensee planned maintenance activities to correct the identified operational deficiencies. References used during this inspection are identified in the attachment to this report.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope (1 Sample)

The inspector reviewed permanent modifications to the reactor vessel head vent line accomplished per plant design change (PDC) 03-088, "Installation of 2" Hand Operated Globe Valve on Reactor Vessel Head Vent Line" and 03-077, "Reactor Head Vent Pipe

Coupling Removal." The inspection verified the modification did not degrade the capability of the system as specified in the design and licensing basis or place the plant in an unsafe condition. The review focused on the compatibility of the materials, seismic and structural considerations, impact on operating and emergency procedures, and post maintenance testing requirements. The inspector reviewed the associated 50.59 screening evaluations and verified on a sampling basis that updates to critical drawings and plant procedures had been completed. This review covered one sample.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (6 Samples)

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. References used for this review are listed in the attachment to this report. The review covered the following 6 inspection samples:

- 2.1.8.5 & 2.1.8.3, Reactor Pressure Test following repair of Reactor Vessel nozzle N10 (MR 03117491), 10/8/03
- 8.M.1-32.3, ATS Test After B1 Relay Replacement per MR P9700915, 11/4/03
- 8.M.2-2.1.11, Emergency Buses A5 & A6 4.16KV Startup Transformer Undervoltage and Degraded Voltage Relays following replacement of relay 127-504Y per MR 03119757, 11/14/03.
- 2.1.8.4, Repair (Non-Safety Class) and Fabricated Replacement (Safety or Non-Safety Class) Pressure Tests Procedure for work performed on the reactor vessel head vent line under MRs 03109752, 0311074, 03113124 and 03116246.
- 2.1.8.4, Repair (Non-Safety Class) and Fabricated Replacement (Safety or Non-Safety Class) Pressure Tests Procedure for work performed on the reactor vessel head vent line under the MR 03117342.
- MR 03118567, MSIV AO-203-2A packing adjustment.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope (1 sample)

Review of Outage Plan

The inspector reviewed the risk profile for the October forced outage 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. Daily risk up-dates, accounting for schedule changes and unplanned activities were also periodically reviewed.

Review of Plant Shutdown and Cooldown Activities

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," were reviewed. Portions of various activities to place the plant in a cold shutdown condition on shutdown cooling were observed by the inspector. The inspection focused on assessing 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. The inspector's observations concerning the implementation of procedure 2.1.7 were discussed with operations management and captured in condition reports 2003-04544 and 2004-00043.

Review of Control Rod Drive 30-35 Activities

The inspector reviewed licensee activities to unlatch and remove the drive mechanism for control rod 30-35 during the outage, which had exhibited anomalous operation when moved with the reactor manual control system since startup for Cycle 15. The inspector reviewed licensee controls over core reactivity and shutdown margin during the maintenance. The licensee deferred efforts to replace the drive during a subsequent outage when it was determined that the drive could not be confirmed to be unlatched. The licensee inserted and deactivated control rod 30-35 at position 00 for subsequent reactor operations. Licensee actions to address control rod 30-35 operability are also discussed in Sections 1R04 and 1R15 of this report.

Review of Drywell Repair Activities for RCS Leakage and Reactor Vessel Nozzle N10

The inspectors reviewed on an ongoing basis the licensee's efforts to monitor, trend and identify sources of unidentified leakage inside the drywell. Identified and unidentified drywell leakage rates and their rates of change were regularly compared to Technical Specification requirements and administrative limits. Trouble shooting plans developed to help determine the source and potentially minimize its effect were reviewed. Inspection of accessible areas within the dry well were performed while shutdown to identify potential leakage sources. The licensee's actions for the reactor coolant system (RCS) pressure boundary leakage, identified in reactor vessel (RV) nozzle N10 cap-to-nozzle weld, were reviewed (Condition Report 200303719). The inspection of the N10 activities focused on the following areas:

- characterization of the flaw and the mechanism which facilitated the crack growth through wall
- scope expansion in regard to performing additional RV nozzle ultrasonic examinations
- impact of potential repair plans on reactor plant conditions, vessel level, and emergency core cooling system activation logic
- the weld repair procedures and welder qualifications
- post repair testing
- root cause evaluation

The Updated Final Safety Analysis Report, Technical Specifications, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements (Section IX and XI), Generic Letter 88-01 and its supplements, and the Boiling Water Reactor Vessel Integrity Program (BWRVIP-75) document were utilized in the inspection to assess the adequacy of the licensee's actions. Additionally, conference calls involving NRC and licensee personnel were held to assess the adequacy of the flaw evaluation, the structural weld overlay repair and planned testing, and the scope expansion determination detailed in an engineering evaluation dated October 7, 2003. A listing of specific documents reviewed is provided in the attachment to this report.

b. Findings

Introduction

On October 1, 2003, during performance of drywell inspections to identify the source(s) for the increased trend in unidentified drywell leakage, a reactor coolant system (RCS) pressure boundary leak was identified in reactor vessel (RV) nozzle N10 cap-to-nozzle weld. The pressure boundary leak was self-revealing and determined to be of very low safety significance. Plant operation with pressure boundary leakage is a violation of Technical Specification requirement 3.6.C, however Enforcement Discretion is warranted.

Description

N10 is a 4 inch RV penetration located approximately 84 inches above the top of active fuel and was previously used to return control rod drive (CRD) system flow to the RV. The CRD line was removed and the penetration capped in November of 1977. Ultrasonic examination (UT) of the leak area identified a circumferential flaw in the weld material 1.75" in length with a semi-elliptical depth profile that started out shallow at the ends and propagated radially toward the center of the flaw where it leaked through. The licensee's root cause analysis concluded the through-wall crack was caused by an incipient crack or crevice condition which remained in the weld after repair welding performed as part of the nozzle-to-cap fabrication in 1977. Subsequent crack propagation continued through-wall by interdendritic stress corrosion cracking (IDSCC) mechanism due to high residual weld stresses in the inconel 82/182 weld metal as a result of the repair. The embryonic crack caused by the repair welding was not detected by the final fabrication radiographs (RT) and preservice UT inspections performed in 1977. Once the crack developed and began to propagate, the flaw remained undetected during subsequent periodic inspections. The nozzle was last inspected by manual ultrasonic testing in 1999, at which time no crack-like indications were noted. Leakage from N10 was not evident during the May 2003 in-service pressure test. However, a bare metal visual inspection of the nozzle, which is normally insulated, was not accomplished or required during this test.

The purpose of the ISI program is to detect flaws/cracks before they become a significant defect. It did not do so which is indicative of a failure of the Inservice Inspection (ISI) Program to perform its function. This condition has been addressed through industry and NRC initiatives to improve hardware, technology, training standards, and personnel qualification requirements. The failure to identify the flaw in prior inspections does not constitute a licensee performance deficiency. The flaw went undetected despite examination using the best available equipment and methods at the time and the examinations were performed in accordance with the existing regulatory and ASME Code requirements.

Operation of the plant with reactor coolant system pressure boundary leakage is prohibited by Technical Specification 3.6.C.1.a.1 and constitutes a violation of TS requirements. Though Entergy did not identify the N10 pressure boundary leak until October 1, and met the associated TS action statement (TS 3.6.C.1.d) from the point of discovery, it is reasonable to conclude that the pressure boundary leak existed for some in-determinant period of time prior to discovery and during plant operations contrary to the requirements of TS 3.6.C.1.a.1, which prohibits plant operation with pressure boundary leakage. However, this fact does not in itself constitute a performance deficiency. Unidentified drywell leakage was a focus of station management. The increasing trend was closely monitored and operations often increased the monitoring frequency to ensure that changes in both the leak rate and rate of change were appropriately captured and evaluated against TS and administrative requirements. Drywell unidentified and identified leakage remained well within TS limits; including the more conservative station administrative limits, throughout the entire operating cycle. In addition to aggressively monitoring the leak rate via dry well sump pumpdowns, atmospheric and liquid samples were routinely analyzed for trends and diagnostic and confirmatory information. A conservative action to shutdown and identify the source of the unidentified leakage was taken, which led to the discovery of pressure boundary leakage on October 1.

Analysis

During the safety significance assessment, which was qualitative in nature, the inspectors focused on two primary criteria which were (1) whether the leakage condition, based on the characteristics of the mechanism that caused the leak, could have reasonably resulted in an increase in the loss of coolant accident (LOCA) frequency, and (2) whether the leakage could have resulted in the failure of mitigating systems to perform their intended safety function. Based on the characteristics of the N10 flaw, it was concluded that the leak would not reasonably have resulted in an increase in the LOCA frequency. It was also concluded that the leakage would not result in the failure of mitigating systems to perform their intended safety function. Additionally, the inspectors determined that the licensee had mitigating procedures, routine inspection activities, operable leakage detection equipment and TS requirements designed to detect low levels of leakage from the RCS and minimize the potential that a flaw could remain undetected. Based on the above, the inspectors determined, qualitatively, that the N10 pressure boundary leakage was of very low safety significance.

Enforcement

The inspector concluded that the RCS pressure boundary leak resulted from an equipment failure that was not avoidable by the implementation of reasonable quality measures or management controls. The inspector further concluded that the licensee had appropriately identified and corrected the condition, taken action to implement measures to prevent recurrence, and adequately characterized the extent of condition and safety significance. Accordingly, although RCS pressure boundary leakage is a violation of NRC requirements, the NRC has decided to exercise enforcement discretion in accordance with VII.B.6 of the NRC Enforcement Policy and refrain from issuing enforcement action for the violation. EA 04-021

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (4 Samples)

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 review covered the following 4 inspection samples:

- 3.M.3-47, Load Shed Relay Operational/Functional Test (Attachment 2 - Functional Test of Initiation Circuits Associated with B EDG).
- 8.M.2-2.10.2-7, LPCI Loop Selection Initiation Logic System Test
- 2.1.8.5, Reactor Vessel Pressurization and temperature Control for Class 1 System Leakage Test.
- 8.5.4.1, HPCI Quarterly Testing, December 10, (Condition Report 200304493)

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 Sample)

The inspector observed training of licensed operators to evaluate the operators ability to properly classify plant events in accordance with the Emergency Action Levels and complete the required notifications for plant events.

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 (11 Samples)

During the period from October 6-10, 2003, the inspector reviewed exposure significant work areas, high radiation areas, and airborne radioactivity areas in the reactor (including drywell) 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. Significant radiological work being performed at the time of this inspection included activities associated with a mid-cycle outage which included nozzle (N10) weld repair. 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.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (1 Sample)

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).

Year-to-date and mid-cycle outage exposure goals were compared with results achieved. Year-to-date exposures exceeded estimates both from the spring refueling outage (RFO14) and for the balance of the year, due, in-part, to a number of forced shutdowns, including the mid-cycle outage which is estimated to add 26 person-rem of exposure to the yearly total. Prior to the commencement of the mid-cycle outage, site year-to-date exposures were 216 person-rem.

The corporate exposure goal established for RF014 was 150 person-rem, while pre-outage estimation of the work scope was 196 person-rem. Major jobs during RFO14 included: 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. Post-outage review of the RFO14 exposure goals indicated that exposures were approximately 167 person-rem.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (71121.03)

a. Inspection Scope (1 Sample)

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.

The inspector reviewed the licensee program for utilization of atmosphere supplying suits to meet the rescue requirements of 10 CFR 20.1703(f). Airline supplied respirators are used at Pilgrim during entries under the reactor vessel during control rod drive work.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety (PS)

PS3.1 Radiological Environmental Monitoring Program (REMP) (71122.01)

a. Inspection Scope (6 Samples)

The inspector reviewed the following documents and observed the licensee’s activities to evaluate the effectiveness of the Entergy’s REMP at the Pilgrim site. The requirements of the REMP are specified in the Technical Specifications/Offsite Dose Calculation Manual (TS/ODCM). This inspection activity represents the completion of six (6) samples relative to this inspection area (i.e., inspection procedure sections 02.01a,b,c,d and 02.02a, b, d, e, f, g, h, i).

- the 2001 and 2002 Annual REMP Reports;
- selected analytical results for 2003 REMP samples;
- the most recent ODCM (Revision 9) and technical justifications for ODCM changes, including sampling media and locations for Revision 9;
- the 2003 QA Audit Reports (Audit Report No. 03-07) for the REMP/ODCM implementations;
- review of 2002 QA Surveillance Reports (02-009, 02-041, 02-065 and 02-070);
- the most recent calibration results for all TS/ODCM air samplers;
- review of QA/QC of the contractor laboratory (Framatom ANP Environmental Laboratory and Environmental Laboratory, FitzPatrick Nuclear Power Plant);
- implementation of the environmental thermoluminescent dosimeters (TLDs) program;
- the Land Use Census procedure and the 2002/2003 results;
- associated procedures to implement the REMP;
- observation for air iodine/particulate and water sampling techniques; and,
- walk-down for determining whether all air samplers and 25%TLDs were located as described in the ODCM (including control and indicator stations) and for determining the equipment material condition.

b. Findings

No findings of significance were identified.

PS3.2 Meteorological Monitoring Program (71122.02)

a. Inspection Scope (1 Sample)

The inspector reviewed the following documents and observed the licensee's activities to evaluate the effectiveness of Entergy's Meteorological Monitoring Program. The requirements of the Meteorological Monitoring Program are specified in Section 2.3 of the Updated Final Safety Analysis Report (UFSAR). The monitoring requirements (e.g., calibration accuracies and the annual recovery of meteorological program data in Section 2.3 of UFSAR) commit the licensee to follow Regulatory Guide 1.23 (RG 1.23). This inspection activity represents the completion of one (1) sample relative to this inspection area (i.e., inspection procedure section 02.02c).

- the 2003 quarterly calibration results of the meteorological monitoring instruments for wind direction, wind speed, and temperatures, and calibration procedure;
- 2002 meteorological monitoring data recovery statistics required by RG 1.23 (the required annual data recovery is greater than 90%);
- selected Condition Reports (CRs) [21 CRs for the year 2002 and 22 CRs for the year 2003] to evaluate the effectiveness of the Entergy's problem identification and resolution processes in the area of the Meteorological Monitoring Program;

- observation for the operability of meteorological monitoring instruments at the tower; and
- observation of meteorological instrumentation weekly checks at the tower.

b. Findings

No findings of significance were identified.

PS3.3 Radioactive Material Control Program (71122.03)

a. Inspection Scope (2 Samples)

The inspector reviewed the following records to ensure that the Entergy met the requirements specified in its program for the unrestricted release of material from the Radiologically Controlled Area (RCA). The review was against criteria contained in 10CFR20, NRC Circular 81-07, NRC Information Notice 85-92, NUREG/CR-5569, Health Position Data Base (Positions 221 and 250), and the licensee's procedures. This inspection activity represents the completion of two (2) samples relative to this inspection area (i.e., inspection procedure sections 02.03a and 02.03 b, c, d, e).

- the most recent calibration results for the radiation monitoring instrumentation;
- the survey and release of potentially contaminated bulk material;
- the methods used for control, survey, and release from the RCA; and
- associated procedures.

b. Findings

No findings of significance were identified

4. Other Activities (OA)

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope (5 Samples)

1. Occupational Radiation Safety Cornerstone

The inspector reviewed a listing of licensee event reports for the period January 1, 2003 through September 30, 2003, for issues related to the occupational radiation safety performance indicator, which measures nonconformance with high radiation areas greater than 1R/hr and unplanned personnel exposures greater than 100 mrem TEDE, 5 rem SDE, 1.5 rem LDE, or 100 mrem to the unborn child.

2. Barrier Integrity Cornerstone

The inspector reviewed the reactor coolant system specific activity performance indicator data for the fourth quarter 2002 through the third quarter 2003 to verify the accuracy and completeness of the submitted data. The inspector reviewed NEI 99-02, "Regulatory Assessment Performance Indicator Guideline"; observed a sampling evolution on November 12, 2003; and reviewed past reactor coolant sample result data.

3. Mitigating Systems Cornerstone

a. The inspector reviewed the safety system functional failure performance indicator data for the fourth quarter 2002 through the third quarter 2003. The inspector reviewed licensee event reports, portions of operator logs, maintenance records, maintenance rule documents, and NRC Inspection reports for the period of October 2002 to September 2003 to determine the accuracy and completeness of the reported performance indicator. The inspector verified that the licensee had classified safety system failures in accordance with NRC endorsed criteria contained in NEI 99-02, "Regulator Assessment of Performance Indicator Guideline." The inspector also verified the licensee's program would address anomalies in equipment performance and data reporting.

b. The inspector reviewed the safety system unavailability (SSU) performance indicator (PI) data for the residual heat removal (RHR) system functions of suppression pool cooling (SPC) and shutdown cooling (SDC). The review encompassed portions of the second and third quarter 2002 in addition to the fourth quarter of 2002 and first quarter 2003. The inspector reviewed operator and LCO logs, RHR system tagouts, work orders, and condition reports to assess the completeness and accuracy of the reported PI data. Interviews with the system engineer were also conducted.

4. RETS/ODCM Radiological Effluent Occurrences

The inspector reviewed the following documents to ensure Entergy met all requirements of the performance indicator from the third quarter 2002 to the third quarter 2003: monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases; quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and, associated procedures.

b. Findings

URI. The resident inspectors raised a concern that the licensee misapplied the NEI 99-02 guidance when they calculated the RHR SSU resulting in a non conservative report in the second, third, and fourth quarter 2002. In response to this concern the licensee performed a bounding analysis for the RHR SSU which indicated that the PI will not cross the Green-White threshold. An extent of condition review for potential impact in other PIs is forth coming. The issue was entered into the corrective action program (CR 04-0036). Three underlying issues relating to implementation of NEI 99-02 guidance were identified:

1) determination of SSU time based on after the fact time estimates.

2) determination of SSU time by subtracting out time for individual procedure steps, during which the system was believed to be in a configuration which had no effect on the monitored functions, was not adequately justified nor applied.

3) application of the 99-02 guidance to exclude SSU time based on operator action was incorrectly applied to restoration of the shutdown cooling (SDC) function. The licensee credited operator action for conditions beyond those allowed by NEI 99-02 and without meeting all of the criteria. The licensee's action was based on the ability to align the RHR system for the SDC function, a manual action, within the time period that the SDC mode of RHR would be required.

Entergy is evaluating whether to seek clarification and/or approval via the frequently asked question (FAQ) process. This item is unresolved pending the completion of licensee actions to resolve the above issues through the FAQ process, and subsequent NRC review (**URI 50-293/03-11-02**).

4OA2 Identification and Resolution of Problems (71152)

Effectiveness of Corrective Actions

1. Routine Review of Identification and Resolution of Problems (71152)

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," the inspector 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

No findings of significance were identified.

2. Vessel Partial Drain Down Events (71152)

a. Inspection Scope (1 PI&R sample)

The inspector reviewed the Condition Reports (CRs) and related documentation regarding the three reactor pressure vessel (RPV) partial draindown events that occurred during the year 2003. The inspector conducted this review to ensure that the licensee properly identified and evaluated the problems associated with the events, and implemented appropriate corrective actions. The partial draindowns were previously documented in NRC inspection reports 293/2003-004 (CR 20030651), and 293/2003-006 (CR 20031663, CR 20032010) and are summarized as follows:

On February 13, 2003, an unintended decrease of about 21" in the RPV level occurred during a swap of residual heat removal (RHR) loops providing shutdown cooling because the operators did not fully close valve MO-1001-28A as directed by the procedure. A contributing factor included an unclear procedure.

On April 27, 2003, an inadequate tagout restoration resulted in an unintended drain path from the RPV to the clean radwaste tanks via the control rod drive system. The restoration sequence had been modified without an adequate review of the change prior to implementation.

On May 9, 2003, a draindown of the RPV through the automatic depressurization system (ADS) valves to the torus was initiated by use of an inadequate test procedure and the failure of the staff to recognize that the RPV water level was high enough to cause the ADS valve to unseat.

The inspector reviewed the condition reports for the above three events to assess the adequacy of the problem description, evaluation, and the scope of the corrective actions. In addition, the inspector discussed the events with the Operations Support Superintendent, assistant Operations Manager, and nuclear assessment supervision to determine the status of actions to correct the identified problems, and steps to evaluate the effectiveness of the corrective actions.

b. Findings

No findings of significance were identified.

3. Emergency Diesel Operating in Parallel with Electrical Grid (71152)

a. Inspection Scope (1 PI&R sample)

During the review of industry operating experience in April 2003, the licensee identified potential problems if a single failure occurred while an emergency diesel generator is operating in test parallel with the offsite power supplies (reference Operating Experience OE 14874 and OEN-02-00215). The issue involves the diesel providing a voltage signal to the relays used to monitor the offsite electrical system, and the potential masking of a degraded grid condition. The licensee’s actions to address the issues in CR 200301278 were selected for review because they impact a risk significant plant system.

The inspector reviewed the licensee’s root cause analysis and corrective actions to address this condition to ensure corrective actions commensurate with the significance of the issue have been identified and implemented by the licensee. The inspector determined that the licensee classified the condition appropriately and considered the corrective actions to address the conditions to be reasonable.

The inspector verified selected corrective actions to be complete. The inspector confirmed that the licensee engineering evaluation identified the appropriate corrective actions to address the potential vulnerability; the interim corrective actions were appropriate and implemented as planned; procedures were changed to direct operator response to conditions that potentially degrade the emergency power supply; and, the conclusions in the final engineering evaluation were reasonable to demonstrate the emergency power supply remained operable. The inspector confirmed actions taken to revise surveillance procedures and to enter the technical specification limiting condition for operations when the emergency diesels are tested in parallel with the startup transformer.

b. Findings

No findings of significance were identified.

4. Condition Reports in Radiation Protection (71121)

a. Inspection Scope (1 sample)

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 35 condition reports of issues related to 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.

5. Condition Reports in Effluents (71122)

a. Inspection Scope (1 sample)

The inspector reviewed the following 2002-2003 Condition Reports (CRs) to evaluate the effectiveness of the licensee’s problem identification and resolution processes in the areas of the REMP and the Radioactive Material Control Program. This inspection activity represents the completion of one (1) sample relative to this inspection area (i.e., inspection procedure section 02.04a, b, c).

- CRs for REMP (CR-PNP-2002-13126; CR-PNP-2002-12695; CR-PNP-2002-10976; CR-PNP-2002-00266; CR-PNP-2003-00777; CR-PNP-2003-02006; CR-PNP-2003-02936; CR-PNP-2003-02937; CR-PNP-2003-02938; CR-PNP-2003-03192; CR-PNP-2003-03194; and CR-PNP-2003-03198); and,

- CRs for Radioactive Material Control Program (CR-PNP-2003-03708; CR-PNP-2003-03646; CR-PNP-2003-01342; CR-PNP-2003-00780; CR-PNP-2003-00692; CR-PNP-2002-12887; CR-PNP-2002-12424; CR-PNP-2003-09564; and CR-PNP-2003-09717).

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

1. HPCI Inoperable During Surveillance Test (71153)

a. Inspection Scope (1 sample)

During a surveillance test of the high pressure coolant injection (HPCI) system on August 29, 2003, the operators manually tripped the HPCI system and declared it inoperable when HPCI automatically tripped and reset two times after operating about 25 minutes in the full flow test mode. NRC Inspection Item 2003-07-01 was open pending completion of the NRC review of past HPCI testing and maintenance; the licensee event evaluation and corrective actions; and, a determination of whether HPCI could perform its safety function during plant operations prior to the August 29 test. The licensee reported this event per 10 CFR 50.73(a)(2)(v)(B) and 50.73(a)(2)(v)(D) as Licensee Event Report (LER) 2003-004.

b. Findings

Introduction

Green. The failure to provide adequate guidance in HPCI maintenance procedures resulted in the HPCI system being inoperable for repairs for about 3 days during plant operations. This issue constitutes a self-revealing finding of very low safety significance (Green) and a non-cited violation of Technical Specification (TS) 5.4, "Procedures." The failure to adequately incorporate operating experience in maintenance procedures is a contributing cause in the cross-cutting area of problem resolution.

Description

The HPCI system was tested satisfactorily for a one hour operability run during plant startup from the refueling outage (RFO#14) on May 13, 2003. The licensee started HPCI for a routine test per procedure 8.5.4.1 on August 29, 2003. This was the first HPCI quarterly test since startup from RFO #14. After about 25 minutes of operation in the full flow test mode, the operators manually tripped the system when HPCI automatically tripped and reset two times for unknown reasons. HPCI was declared inoperable at 1:50 p.m. on August 29. Subsequent investigation identified that the turbine stop valve (TSV) stem had failed leaving the valve in the open position and that the hydraulic portion of the mechanical overspeed trip device had malfunctioned to cause the repetitive trips. HPCI remained out of service as the licensee investigated the problems and made repairs. After investigating the failures and completing short term corrective actions, the licensee returned HPCI to an operable status at 6:56 p.m. on September 2, 2003.

The conditions that contributed to HPCI performance during the quarterly test revealed weaknesses in licensee maintenance procedures and practices. The turbine stop valve (TSV) stem failed during the automatic trip/reset cycles during the August 29 test. The stop valve stem failed because an improper balance chamber adjustment resulted in a stem overstress condition. Licensee procedure 3.M.4-81did not reflect current industry practices in setting the balance chamber pressures, which increased the operating stresses on the shaft. Further, a manufacturing deficiency allowed nitrides on the threaded portion of the shaft during the hardening process, which lead to cracks in the shaft. The root cause of the stem failure was the improper balance chamber pressure adjustment.

The inadvertent overspeed trips occurred due to a malfunction of the hydraulic portion of the mechanical overspeed trip device. The overspeed trips occurred even though there was no overspeed condition. Of several factors that impacted the performance of the overspeed trip device, the most significant deficiencies were the improper tension on the reset spring and the improper reset set point on valve PSV-2301-238. The tension on the trip reset spring was set at 1 pound instead of the desired setting of 2 to 5 pounds (reference MR P9402658 and 3.M.4-84 dated 5/12/03). Industry experience indicates the value should be about 5 pounds. The reset set point on PCV-2301-238 was 7.5 psig instead of the desired value of 10 psig. The licensee preventive maintenance procedures did not periodically calibrate or test the setting on PSV-2301-238. The combined effect of these two conditions was that the overspeed tappet would “float” between the trip and reset position when the balance point was reached between the reset spring and the oil pressure under the tappet. The root cause of the HPCI failure was that the spring setting on the overspeed trip device was set too low due to a procedure that lacked sufficient guidance.

NRC review determined that HPCI could have performed its design function if called upon prior to the August test as long as there were no additional failures other than those revealed by the August 29 test and identified during the subsequent licensee investigations. Other deficiencies addressed by the licensee in the corrective action program were identified in Condition Reports 200303302, 200303305 and 200303321. Although the conditions that contributed to the HPCI performance on August 29 existed prior to the test, the malfunction of the overspeed trip device caused HPCI to fail during the test; the failed stop valve stem did not keep HPCI from pumping water. The HPCI deficiencies became apparent when the operators failed to positively reset the tappet after testing the overspeed trip device during the pre-start checks completed just prior to the August 29 surveillance test. The combination of operator knowledge and guidance in Steps [14] and [15] of test procedure 8.5.4.1 were inadequate to assure the overspeed trip device was properly reset. The licensee revised procedure 8.5.4.1 to assure the overspeed device is positively reset during the pre-start checks. However, HPCI was operable following RFO#14 up to the time the pre-start checks were completed on August 29.

<u>Analysis</u>

A test of the HPCI system per 8.5.4.1 is normally completed in less than 12 hours (reference the May 2003 test). The HPCI system was out of service for a total of 89 hours following the August 29 test, or about 77 hours longer than needed for a routine test. The impact of the licensee performance deficiencies was to make HPCI inoperable necessitating repairs during 3 days of reactor power operations.

In accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Disposition Screening,” the inspectors determined that the issue was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone objective. Specifically, the availability of the high pressure coolant injection (HPCI) pump was adversely impacted by inadequate maintenance practices on the overspeed trip device and valve PSV-2301-238. In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted an SDP Phase 1 screening and determined that an SDP Phase 2 evaluation was required because the performance deficiency resulted in an actual loss of safety function of the HPCI system.

The inspectors conducted an SDP Phase 2 evaluation of the risk significance of the performance deficiency and determined that the finding was of low to moderate safety significance (White). The inspectors used the following assumptions in the Phase 2 evaluation.

- The HPCI pump would have failed to fulfill its function, if called upon, for a period of approximately 89 hours. Therefore, an exposure time of between 3 and 30 days was used in the analysis.

- The HPCI pump was not able to be recovered in a time frame to support successful high pressure injection. Therefore, recovery of the HPCI pump was not credited.

The inspectors reviewed the Phase 2 results and concluded that they were conservative for two reasons. First, the failure rates for the reactor core isolation cooling (RCIC) system in the SDP Phase 2 notebook were conservative in comparison to the actual system failure rates. Second, the counting rule conservatively adds the solved Phase 2 accident sequences in a simplified manner. As a result, the inspectors determined that a Phase 3 analysis of this finding was appropriate.

The regional Senior Reactor Analyst conducted the SDP Phase 3 analysis using the assumptions stated above. The analyst used the NRC’s SPAR model, Revision 3.01, to evaluate the significance of this finding which was revised to reflect licensee operating experience with the RCIC system. This change resulted in a revised core damage frequency of 1.56E-5 per year. The analyst determined that the change in core damage frequency for this finding was 9.3E-8 per year. The dominant accident sequence involved a loss of main feedwater initiating event, failure of the power conversion system, failure of the RCIC system, and failure of the operators to depressurize the reactor. As a result, the analyst determined that the inadequate maintenance practices associated with valve PSV-2301-238 in the HPCI system were of very low risk significance (Green).

NRC review noted areas for improvement related to problem resolution. For example, the maintenance procedures that lacked sufficient guidance (for balance chamber pressure and spring tension adjustments) - did not reflect the latest industry recommendations; this highlights a weakness in incorporating industry operating experience in the maintenance program. Further, NRC identified the need for additional corrective actions to identify the source of debris and to prevent internal contamination of the oil system (the actions were later included in CR 200303302). These aspects of the finding are examples of a contributing cause in the cross-cutting area of problem resolution, as described in Section 4OA4 below.

Enforcement

The failure to have adequate maintenance procedures were examples of a violation of Technical Specification 5.4.1. Due to its low safety significance, this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy (NUREG 1600). Unresolved Item 2003-07-01 is considered closed. This issue is addressed in the licensee corrective action program as Condition Report 200303302. (**NCV 50-293/03-11-03**).

2. Licensee Event Report Review and Closeout (71153)

a. (Closed) LER 50-293/2003-004: HPCI Inoperable During Surveillance Test Due to Inadequate Procedure The inspector reviewed the licensee actions associated with Licensee Event Report (LER) 50-293/2003-004. Corrective actions were described in Condition Report 200303302. This event was also described in Section 4OA3.1 above and NRC Report 2003-07. The LER provided an accurate description of the event and the licensee followup actions. The inspector noted that subsequent licensee evaluations had identified that the stop valve stem had failed due to an overstress condition. This LER is closed.

b. (Closed) LER 50-293/2003-006: Reactor Coolant Pressure Boundary Leakage due to Reactor Vessel Nozzle Weld Crack Propagation: The inspector reviewed the corrective actions associated with Licensee Event Report (LER) 50-293/2003-006, corrective actions were described in Condition Report 2003-03719. The event was also described in section 1R20 to this report. The LER provided an accurate description of the event and the licensee followup actions. This LER is closed.

.3 Review of Previous Inspection Items (71153)

(Closed) Unresolved Item URI 2003-07-01: HPCI Operability During Plant Operations Prior to the August 29 Surveillance Test. This item was reviewed as described in Section 4OA3.1 of this report. URI 2003-07-01 is closed.

4OA4 References to Cross Cutting Issues

Section 1R14 describes an example of a human performance error that impacted plant operations. The event occurred despite past and ongoing efforts to improve human performance. This was the second example within the past year where error prevention techniques were not successful in preventing operational events in which time pressures were a contributing factor (reference Inspection Item 2003-06-01).

Section 4AO3.1 describes an NRC finding where the failure to adequately incorporate operating experience in a maintenance procedure resulted in the HPCI system being inoperable during three days of plant power operation. This was an example of a contributing cause in the cross cutting area of problem resolution.

4OA6 Meetings, Including Exit

The inspector summarized NRC inspection results during periodic meetings with licensee management, which included the findings in the area of Problem Resolution presented in a meeting on December 11, the Radioactive Effluents area on October 23, and the Radiological Controls area on September 12.

The inspector presented a summary of inspection results to the licensee at the conclusion of the inspection on January 8, 2004. The licensee acknowledged the conclusions and observations presented. The inspector discussed the materials examined during the inspection that should be considered proprietary, and none were included in the inspection findings.

4OA7 Licensee-Identified Violation

The following violation of very low safety significance (Green) was identified by the licensee 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 NCV.

Section 2.3 of Updated Final Safety Analysis Report (UFSAR) requires the licensee to maintain the onsite Meteorological Monitoring Program as specified in Regulatory Guide

1.23 (RG 1.23). The required percentage for annual data recovery per RG 1.23 is greater than 90%. The licensee's data recovery percentage for the entire year of 2002 was about 77% for the 33-ft level, and 63% for the 220-ft level of the meteorological tower. The licensee reported this failure in the 2002 Radiological Effluent and Waste Disposal Report as required by the TS/ODCM. The licensee had identified corrective actions (CR-PNP-2003-03975 and MET Tower System Reliability Improvement Plan) to enhance the system, including the option for the installation of new monitoring systems.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Burke	Security Operations Supervisor
P. Dietrich	General Manager, Acting Vice President
V. Fallacara	Training Manager
B. Ford	Licensing Manager
L. Lapeyrouse	Medical/Access Authorization
W. Lobo	Licensing Engineer
W. Mauro	ALARA Supervisor
D. Noyes	Assistant Operations Manager
D. Perry	Radiation Protection Manager
R. Rose	Security Manager
M. Santiago	Operations Training Superintendent
T. Sowdon	Emergency Preparedness Manager
E. Solomon	Emergency Preparedness Planner
S. Willoughby	Operations Training Supervisor

NRC

Stephanie Coffin	Section Chief, Materials and Chemical Engineering Branch, Division of Engineering
Terence Chan	Section Chief, Materials and Chemical Engineering Branch, Division of Engineering
Andrea Lee	Materials and Chemical Engineering Branch, Division of Engineering
William Koo	Materials and Chemical Engineering Branch, Division of Engineering
James Clifford	Section Chief, Division of Licensing Project Management
William Beckner	Branch Chief, Reactors Operations Branch, Division of Inspection Program Management
Jerry Dozier	Reactors Operations Branch, Division of Inspection Program Management
Samuel Lee	Reactors Operations Branch, Division of Inspection Program Management

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Open and Closed

050002932003011-01	NCV	Technician failed to follow procedures causing trip of B recirculation pump and entry into single loop operations

05000293/2003011-03	NCV	Failure to Implement Adequate Procedures Caused HPCI to be Inoperable for Maintenance for 3 days.

Open

05000293/2003011-02	URI	Reporting of RHR SSU in accordance with NEI 99-02

Closed

05000293/2003007-01	URI	HPCI Operability During Plant Operations Prior to the August 29 Surveillance Test.
05000293/2003-004	LER	HPCI Inoperable During Surveillance Test due to Inadequate Maintenance Procedure
05000293/2003-00600	LER	Reactor Coolant Pressure Boundary Leakage due to Reactor Vessel Nozzle Weld Crack Propagation, EA 04-021

LIST OF DOCUMENTS REVIEWED

References for Section 1R04

Clearance 1-02-1, Tagout 03-0054 for HCU 30-35
Technical Specifications 3.3.B, “Control Rod Operability”
Technical Specification 3.9.B, “Auxiliary electrical System”
UFSAR Section 3.4, Reactivity Control Mechanical Design
Procedure 2.1.12.1, Emergency Diesel Generator Daily Surveillance
UFSAR Sections 8.1, 8.2 and 8.4, Electrical Power Systems
Technical Specifications 3.9.A and 3.9.B, Auxiliary Electrical Systems
Technical Specification 3.5.C, High Pressure Coolant Injection
UFSAR Sections 6.4.1 and 6.5.2.3, High Pressure Coolant Injection System
HPCI Operational Surveillance Test per Procedure 8.5.4.1, dated 5/13/03, 8/29/03 and 9/2/03
Procedure 2.2.21, High Pressure Coolant Injection System
Engineering Evaluation 03-014, Condensate Storage Tank Coating
Report 89XM-1-ER-Q-E5, Updated Fire Hazards Analysis
Procedure 2.2.21.5, HPCI Injection and Pressure Control
System Health Report for HPCI
Drawing 243, HPCI System
Drawing 244, HPCI System, sheets 1 & 2

References for Section 1R05

Report 89XM-1-ER-Q, Updated Fire Hazards Analysis
Procedure 5.5.2, Special Fire Procedure
Problem Report 01.09665, Deficiency in Appendix R Fire Barrier
Work Request No. 059639, Appendix R Silco Cloth Tear
FP Engineering Evaluation 129, Fire Wrap system with Degraded Silco Cloth
50.59 Screening Review Form dated 7/31/02 for FPEE-129

References for Section 1R13

Temporary Procedure 03-036, Control Rod 30-35 Withdrawal and Removal
Temporary Procedure 03-037, Control Rod 30-35 Removal and Installation

References for Section 1R15
Operability Evaluation for Condition Report 200304081
NEDE-30921-5-P, GE BWR Control Rod Lifetime Limit, Table 3-2
GNF Analysis Report REK-ENN-HK1-03-050
3D Monitor Predictions for Control Rod Depletion
NRC 1985 SER on Amendment 7 to GESTAR II
UFSAR Sections 3.2, 3.4, 3.6
NEA 03-115, SIMULATE-3 Verification of Evaluation of Cycle 15 Stuck Rod
NEA 03-118, CASMO-4 Evaluation of DuraLife-160 Worth Degradation

References for Section 1R16
Procedure 1.3.34.4, Compensatory Measures (CM)
Operator Compensatory Measure Log
CM Evaluation #248, D9 Enclosure High Temperature
CM Evaluation #251, Startup Transformer Temperature Elements Inoperable
CM Evaluation #253, Main Steam Tunnel Temperatures (AO-203-2A Leakage)
Maintenance Request 03113426, Startup Transformer Temperature Switches
Condition Report 200303616, Startup Transformer X4
Condition Report 200303957, Transformer X58 Voltages
Condition Report 200303988, Main Generator Voltage Regulator

References for Section 1R19
Procedure 2.1.8/3, Visual Examination for Leakage During System Pressure Testing, Revision 11, for Test Conducted October 9, 2003
Procedure 2.1.8.5, Reactor Vessel Pressurization and Temperature Control for Class 1 System Leakage Test, Revision 12, for Testing Conducted on October 8-9, 2003
ASME Code Case N-416-2, Alternative Pressure Test Requirement for Welded Repairs, Fabrication Welds for Replacement Parts and Piping Assemblies, or Installation of Replacement Items by Welding, Class 1, 2 and 3
Pilgrim Relief Request (PPR)-36, Alternative Contingency Repair Plan for Generic Letter 88-01, Reactor Pressure Vessel Nozzle-to-End Cap Weld, Using ASME Code Cases N-638 and N-504-2 with Exceptions, Letter 2.03.120 dated October 8, 2003
MR 0311074, Replacement Valves for 4-HO-48 & 4-HO-49
MR 03113124 Child MR To Fabricate Valves in Shop For MR 03109752
MR 03116246 Requesting Section 11 Repair Plan for 4
MR 03117342, Install 2" Globe Valve on Reactor Vessel Head Vent Line

References for Section 1R20
Code Cases:
N-504-2, Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping, Section XI, Division 1 (March 12, 1997)
N-638, Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI, Division 1 (September 24, 1999)

N-416-2, Alternative Pressure Test Requirements for Welded Repairs, Fabrication Welds for Replacement Parts and Piping Subassemblies, or Installation of Replacement Items by Welding, Class 1, 2, and 3.

Weld Procedure Specification

WPS 43-43-S-001 R1 Manual Shielded Metal Arc Welding of P43 to P43
WPS 01-43-S-001 R0 Manual Shielded Metal Arc Welding of P1 to P43
WPS 03-43-T-801 R0 Automatic Machine Gas Tungsten Arc Welding of P3 Group 3 to P43

Welding Procedure Qualification Record

PQR M04343, Manual Gas Tungsten Arc Welding of P43 to P43
PQR 01-43-5-001 Manual Shielded Metal Arc Welding of P1 to P43
PQR 03-03-T-801 Automatic Machine Gas Tungsten Arc Welding of P3 Group 3 to P3 Group 3
PQR A43256-52 R2 Automatic Machine Gas Tungsten Arc Welding of P43 to P43

NDT Data Sheets

ENN-NDE-9.19 R0, Linearity Data Sheet
CDS 03-C-398, Ultrasonic Calibration Data Sheet
CDS 03-C-396, Ultrasonic Calibration Data Sheet
EDS 03-E-398, Ultrasonic Indication Sizing Data Sheet, Weld 3-I-1
EDS 03-E-397, Ultrasonic Examination Indication Report Sheet, Weld 3-I-1
N10 Nozzle Dissimilar Metal Weld 3-I-1 Ultrasonic Examination (10-04-03)

Miscellaneous

PDC 03-089, Weld Overlay for the CRD Return Nozzle N10 to Cap Weld
CR-PNP-2003-03719, CRD Return Nozzle N10 Cap Has Thru Wall Pressure Boundary Leakage
Pilgrim Relief Request (PRR)-36, Alternative Contingency Repair Plan for Generic Letter 88-01, Reactor Pressure Vessel Nozzle-to-End Cap Weld, Using ASME Code Cases N-638 and N-504-2 with Exceptions.
Pilgrim Relief Request (PRR)-36, Alternative Contingency Repair Plan for Generic Letter 88-01, Reactor Pressure Vessel Nozzle-to-End Cap Weld, Using ASME Code Cases N-638 and N-504-2 with Exceptions - Revision 1.
Pilgrim Relief Request (PRR)-37, Alternative Contingency Repair Plan for Generic Letter 88-01, Reactor Pressure Vessel Nozzle-to-End Welds, Using ASME Code Cases N-638 and N-504-2 with Exceptions.
Pilgrim Relief Request (PRR)-38, Relief from ASME Code, Section XI, Appendix VIII, Supplement 11, "Qualification Requirements for Full Structural Overlaid Wrought Austenitic Piping Welds."
GE RICSIL 082, October 17, 1997 Core Spray Nozzle-to-safe end weld leak
GE SIL 455 Revision 1, dated February 22, 1988; Recommendation for Additional ISI of Alloy 182 Nozzle Weldments.
GE SIL 455 Revision 2, dated January 29, 2001; ISI of Additional Alloy 182 Welds.
N10 Nozzle Dissimilar Metal Weld 3-I-1, Ultrasonic Exam, CRD System Return Nozzle and Nozzle Cap Assembly
Engineering Technical Evaluation (10-7-03), Historical Technical Information on Weld 3-I-1
EPRI Summary Report of previous examination history on similar welds (Industry Wide)
NRC Letter dated May 14, 2002; Final Safety Evaluation of the "BWRVIP Vessel and Internals Project, BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)," EPRI Report TR-113932, October 1999(TAC No. MA5012)
EPRI Report TR-113932, October 1999, BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)," - **Proprietary**.

EPRI Report 1008007 Technical Update, October 2003; Dissimilar Metal Piping Weld Examination - Guidance and Technical Basis for Qualification

References for Section 40A2

Condition Reports
CR 20030651, CR 20031663 and CR 20032010.

Other
Corrective Program Event Tree, Work Process, CR Type and Problem Identification Codes used for trending corrective action items.

Procedures
Procedure No. 2.2.19.1, Rev. 9. RHR System - Shutdown Cooling Mode of operation
Procedure No. 8.M.3-1, Rev. 35. Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transfer with Simulated Loss of Off-site Power.
Procedure No. 1.4.5, Rev. 62. PNPS Tagging Procedure

LIST OF ACRONYMS

ALARA	As Low As Reasonable Achievable
BPWS	Banked Position Withdrawal Sequence
CFR	Code of Federal Regulations
CM	compensatory measures
CR	Condition Reports
CRD	Control Rod Drive
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
ICM	Interim Compensatory Measures
IR	Inspection Report
ISI	Inservice Inspection
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MR	Maintenance Request
NCV	Non-Cited Violations
ODCM	Offsite Dose Calculation Manual
OE	Operability Evaluations
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PNPS	Pilgrim Nuclear Power Station
QA	Quality Assurance
REMP	Radiological Environmental Monitoring Program
RFO	Refueling Outage
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RV	Reactor Vessel
RWP	Radiation Work Permit

SDC	Shutdown Cooling
SDP	Significant Determination Process
SSU	Safety System Unavailability
TLD	Thermoluminescent Dosimeters
TS	Technical Specifications
TSV	Turbine Stop Valve
UFSAR	Updated Final Safety Analysis Report