UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

May 4, 2010

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

SUBJECT: PILGRIM NUCLEAR POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000293/2010002

Dear Mr. Bronson:

On March 31, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Pilgrim Nuclear Power Station (PNPS). The enclosed inspection report documents the results, which were discussed on April 15, 2010, with you and members of your staff.

The inspection examined activities performed 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 NRC identified findings of very low safety significance (Green). These two findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs), consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV, 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 DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at PNPS. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Senior Resident Inspector at PNPS. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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Sincerely,

Donald E. Jackson

Donald E. Jackson, Chief
Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 05000293/2010002
w/Attachment: Supplemental Information

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Sincerely,

/RA/

Donald E. Jackson, Chief
Projects Branch 5
Division of Reactor Projects

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

Enclosure: Inspection Report 05000293/2010002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	05000293/2010002
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Pilgrim Nuclear Power Station (PNPS)
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	January 1, 2010 through March 31, 2010
Inspectors:	M. Schneider, Sr. Resident Inspector, Division of Reactor Projects (DRP) B. Smith, Resident Inspector, DRP R. Rolph, Health Physicist, Division of Reactor Safety (DRS) K. Young, Regional Inspector, DRS W. Schmidt, Senior Reactor Analyst, DRS
Approved By:	Donald E. Jackson, Chief Projects Branch 5 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000293/2010002; 01/01/2010-03/31/2010; Pilgrim Nuclear Power Station; Plant Modifications and Post-Maintenance Testing.

The report covered a three-month period of inspection by resident and region based inspectors. Two Green findings were identified, which were determined to be non-cited violations (NCVs). 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 cross-cutting aspect for the finding was determined using IMC 0310, "Components Within The Cross-Cutting Areas," dated February 2010. Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Mitigating Systems

Green. The NRC identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for Entergy's failure to promptly correct a condition adverse to quality. Specifically, Entergy did not correct defective material in their "A" Emergency Diesel Generators (EDG) in a prompt manner which led to emergent maintenance and additional unplanned unavailability of the "A" EDG while they replaced cracked snubber valves. Entergy's corrective actions include entering this issue into the corrective action program and replacing the seven remaining snubber valves on their "A" EDG with those of a material properly hardened and not susceptible to the same mode of cracking.

The inspectors determined that the finding was more than minor because the finding was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone's objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the "A" EDG was unavailable during snubber valve replacements. The inspectors determined the significance of the finding using IMC 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding was determined to be of very low safety significance (Green) because the finding did not result in a loss of system safety function of a single train for greater than its Technical Specifications outage time, and did not screen as potentially risk significant due to external initiating events. This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program component, because Entergy did not take corrective actions in a timely manner. Specifically, Pilgrim did not replace the "A" EDG snubber valves in a prompt manner after repeated fuel leaks from cracked snubber valves over the previous two years. [P.1(d)] (Section 1R19)

Cornerstone: Barrier Integrity

Green. The NRC identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for Entergy's failure to accomplish procedures prescribed for activities affecting quality. Specifically, Entergy did not implement their operability determination process or their temporary modification process for compensatory measures needed to maintain the secondary containment operable. Entergy's corrective actions included designating the compensatory measures as necessary to maintain operability for both torus troughs and implementation of temporary modifications for the equipment installed in the plant to support these compensatory measures.

The inspectors determined that the finding was more than minor because the finding was associated with the Human Performance attribute of the Barrier Integrity cornerstone, and adversely affected the cornerstone's objective to provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, operations and engineering personnel did not adequately implement operability determination and temporary modification procedures when degraded and/or non-conforming conditions associated with the secondary containment torus troughs were identified. The inspectors determined the significance of the finding using IMC 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding was determined to be of very low safety significance (Green) because the finding only represented an impact to the radiological barrier function provided by secondary containment and the standby gas treatment system. This finding had a cross-cutting aspect in the area of Human Performance, Work Practices component, because Entergy personnel did not follow procedures. Specifically, Entergy did not implement their operability determination or temporary modification procedures for compensatory measures needed to maintain the secondary containment operable. [H.4(b)] (Section 1R18)

Other Findings

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

REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station (PNPS) began the inspection period operating at 100 percent reactor power. On March 10, 2010, operators reduced power to 46 percent for a backwash of the main condenser due to a storm surge the previous week. Pilgrim returned to 100 percent reactor power later the same day. On March, 15, 2010 operators reduced power to 93 percent for a control rod pattern adjustment and returned to 100 percent power the same day. Operators maintained the reactor at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Impending Storm

a. Inspection Scope (1 sample)

On the morning of January 25, 2010, a significant winter storm was tracking to impact the Pilgrim plant. The inspectors reviewed Entergy's preparations for the high winds expected to accompany the storm. The inspectors reviewed Entergy's severe weather procedures including; operations during severe weather, coastal storm preparation, and high winds procedures. The inspectors performed a tour of the plant grounds and the switchyard to determine if loose debris or other material could become airborne in the presence of high winds and thereby potentially impact safety related equipment. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns (71111.04Q)

a. Inspection Scope (4 samples)

The inspectors performed four partial system walkdowns during this inspection period. The inspectors reviewed the documents listed in the Attachment to determine the correct system alignment. The inspectors performed a partial walkdown of each system to determine if the critical portions of the selected systems were correctly aligned in accordance with these procedures and to identify any discrepancies that may have had an effect on operability. The walkdowns included selected control switch and valve position checks, and verification of electrical power to critical components. Finally, the inspectors evaluated other elements, such as material condition, housekeeping, and component labeling. The following systems were reviewed based on their risk significance for the given plant configuration:

- Standby Liquid Control System following surveillance testing of both trains;
- Core Spray "A" when Core Spray "B" was out of service;
- High Pressure Coolant Injection following an extended maintenance window; and
- "A" Emergency Diesel Generator following maintenance.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdowns (71111.04S)

a. Inspection Scope (1 sample)

The inspectors completed a detailed review of the "B" Residual Heat Removal (RHR) system to assess the functional capability of the system. The inspectors performed a walkdown of the system to determine whether the critical components, such as valves, breakers, and control switches, were aligned in accordance with operating procedures and to identify any discrepancies that could have an effect on operability. The inspectors discussed system health with the system engineer and performed a review of outstanding maintenance work orders to determine whether the deficiencies significantly affected the "B" RHR system function. The inspectors also reviewed recent condition reports to determine whether "B" RHR equipment problems were being identified and appropriately resolved. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Fire Protection - Tours (71111.05Q)

a. Inspection Scope (5 samples)

The inspectors performed walkdowns of five fire protection areas during the inspection period. The inspectors reviewed Entergy's fire protection program to determine the specified fire protection design features, fire area boundaries, and combustible loading requirements for the selected areas. The inspectors walked down these areas to assess Entergy's control of transient combustible material and ignition sources. In addition, the inspectors evaluated the material condition and operational status of fire detection and suppression capabilities and fire barriers. The inspectors then compared the existing condition of the areas to the fire protection program requirements to determine whether all program requirements were met. The documents reviewed during this inspection are listed in the Attachment. The fire protection areas reviewed were:

- Fire Area 1.9, Fire Zone 1.6, Control Rod Drive (CRD) Pump Quadrant;
- Fire Area 1.9, Fire Zone 1.8, CRD Pump Quadrant Mezzanine Level;
- Fire Area 1.9, Fire Zone 1.15, Standby Liquid Control Pumps and Equipment;
- Fire Area 1.9, Fire Zone 1.16, Open Area – North Side of 91 foot elevation; and
- Fire Area 1.10, Fire Zone 1.2, "B" Residual Heat Removal and Core Spray Quadrant.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

Internal Flooding Inspection

a. Inspection Scope (1 sample)

The inspectors walked down the "B" Residual Heat Removal Quadrant, and associated flood propagation pathways, to assess the effectiveness of Entergy's internal flood control measures. The inspectors assessed the condition of floor drains, walls, and doors. The inspectors also evaluated whether potential sources of internal flooding were analyzed.

b. Findings

No Findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope (1 sample)

The inspectors reviewed one sample of Entergy's program for maintenance, testing, and monitoring of risk significant heat exchangers (HXs) to assess the capability of the HXs to perform their design functions. The inspectors assessed whether the HX program conformed to Entergy's commitments at Pilgrim related to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." In addition, the inspectors evaluated whether potential common cause heat sink performance problems could affect multiple HXs in mitigating systems or result in an initiating event. Based on risk significance and prior inspection history, the "A" Residual Heat Removal Heat Exchanger was selected for detailed review by the inspectors.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope (1 sample)

The inspectors observed licensed operator performance during an emergency planning drill on February 24, 2010. The inspectors observed crew response to a hostile action based scenario which included a loss of all service water. The inspectors assessed the licensed operators' performance to determine if the training evaluators adequately addressed observed deficiencies. The inspectors reviewed the applicable training objectives from the scenario to determine if they had been achieved. In addition, the inspectors performed a simulator fidelity review to determine if the arrangement of the simulator instrumentation, controls, and tagging closely paralleled that of the control room.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope (3 samples)

The inspectors reviewed the three samples listed below for items such as: (1) appropriate work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65 paragraph (b) of the Maintenance Rule; (4) characterizing reliability issues for performance; (5) trending key parameters for condition monitoring; (6) charging unavailability for performance; (7) classification and reclassification in accordance with 10 CFR 50.65 paragraph (a)(1) or (a)(2); and (8) appropriateness of performance criteria for structures, systems, and components (SSCs)/functions classified as paragraph (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified as paragraph (a)(1). The documents reviewed during this inspection are listed in the Attachment. Items reviewed included the following:

- Emergency Lighting Units for Appendix R;
- Functional Failure Determination for Secondary Containment Inoperable due to loss of Torus Trough Water Level; and
- Functional Failure Determination for Reactor Building Closed Cooling Water broken bolt on suction header.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (6 samples)

The inspectors evaluated six maintenance risk assessments for planned and emergent maintenance activities. The inspectors reviewed maintenance risk evaluations, work schedules, and control room logs to determine if concurrent maintenance or surveillance activities adversely affected the plant risk already incurred with out-of-service components. The inspectors evaluated whether Entergy took the necessary steps to control work activities, minimized the probability of initiating events, and maintained the functional capability of mitigating systems. The inspectors assessed Entergy's risk management actions during plant walkdowns. The inspectors reviewed the conduct and adequacy of maintenance risk assessments for the following maintenance and testing activities:

- Planned Yellow Risk for Reactor Core Isolation Cooling Testing;
- Planned Yellow Risk for Load Shed Testing with the Turbine Auxiliary Oil Pump Out of Service;
- Emergent Yellow Risk with "B" Emergency Diesel Generator Out of Service and "A" Residual Heat Removal Pump Out of Service;
- Emergent Green Risk for Inoperable Reactor Building Closed Cooling Water Train;
- Emergent Yellow Risk with the "A" Emergency Diesel Generator Out of Service; and
- Planned Yellow Risk for the Testing of Breaker 504 to the Startup Transformer.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (6 samples)

The inspectors reviewed six operability determinations associated with degraded or non-conforming conditions to determine if the operability determination was justified and if the mitigating systems or barriers remained available such that no unrecognized increase in risk had occurred. The inspectors also reviewed compensatory measures to determine if the compensatory measures were in place and were appropriately controlled. The inspectors reviewed Entergy's performance against related Technical Specifications and Updated Final Safety Analysis Report requirements. The documents reviewed during this inspection are listed in the Attachment. The inspectors reviewed the following degraded or non-conforming conditions:

- CR-PNP-2010-0223, Received PT fuse failure for the "B" Emergency Diesel Generator Relay 160-609;
- CR-PNP-2010-0229, Rod Block Monitor "A" received Rod Block due to loss of input signal;
- CR-PNP-2010-0572, B-18 Motor Control Center temperature is high;

- CR-PNP-2010-0014, Basis for CR-PNP-2009-5295 (Torus Trough Leakage) does not appear to encompass existing conditions;
- CR-PNP-2010-0247, Anticipated transient without Scram Suppression Pool temperature limit higher than previously analyzed; and
- CR-PNP-2010-0063, Received RCIC pump suction pressure high alarm.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Modification Review of Torus Trough Level Indication Installed to Support Compensatory Measures to Maintain Secondary Containment Operability

a. Inspection Scope (1 sample)

The inspectors reviewed the installation of a temporary torus trough level Indication which had been installed to support secondary containment operability. This temporary level indication was implemented in place of system level switches which were determined not to meet design basis requirements for secondary containment operability. The inspectors reviewed condition reports, operability evaluations, and the temporary modification procedure to determine if the system modification should be considered a temporary modification and managed as such. In addition, the inspectors reviewed the system modification and design basis documents to ensure the secondary containment function was not adversely affected.

b. Findings

Introduction: The NRC identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for Entergy's failure to accomplish activities affecting quality in accordance with prescribed procedures. Specifically, Entergy did not implement their operability determination process nor their temporary modification process for compensatory measures needed to maintain the secondary containment operable.

Description: On December 22, 2009, a design engineer conducting a walkdown of the torus room noted that one of the torus troughs (the "A" reactor auxiliary bay floor sump trough) was dry and that the other trough was low in water level. The torus troughs are designed such that reactor auxiliary bay floor sump piping that penetrates the secondary containment are directed to the torus troughs and covered with water to provide a water seal and ensure secondary containment integrity. The engineer notified the control room and operators entered Technical Specification (TS) 3.7.C, Secondary Containment, refilled both torus troughs, and exited the TS. Entergy determined that a low water level switch in the "A" trough had malfunctioned resulting in a failure to receive an alarm in the control room. When this low level alarm is received in the control room, operators are directed by procedures to refill the affected trough in order to maintain secondary containment integrity. On December 30, 2009, in response to the inoperable level switch, Entergy implemented compensatory measures to install a remote camera to monitor the water level in the "A" torus trough until the level switch could be repaired.

On December 31, 2009, the inspectors reviewed the Entergy's corrective actions for the torus trough issue. The inspectors noted that while an operator could observe water in the torus trough using the remote camera, there was no level indicating device to identify when the trough water level was too low to provide a seal and ensure secondary containment integrity. Operators subsequently installed level indicating devices (ruler strapped to the reactor auxiliary bay sump piping); however, they did not designate these compensatory measures as being specified to maintain operability. Requirements associated with compensatory measures to maintain operability are discussed in EN-OP-104, Revision 4, "Operability Determination Process," in Sections 5.4(1), (9), and (10), Section 5.5(6), and Section 5.6. These sections describe the need to identify when compensatory measures are required to maintain operability, to review for the applicability of other processes, such as the temporary modification process, that may be affected, to assess whether the compensatory measures can impact other plant equipment or procedures, and the need to periodically review these compensatory measures to maintain awareness and to ensure timely corrective actions. The inspectors questioned why the torus trough monitoring was not designated as a compensatory measure to maintain operability; however, the "A" torus trough level switch was subsequently repaired on January 21, 2010, and available to warn operators of lowering trough water level. Operators, however, continued to monitor the 'A' torus trough water level using the remote camera and ruler.

On January 27, 2010, design engineering determined that the existing torus trough low water level switch setpoint was not adequate to ensure that secondary containment design requirements were met. Given the conclusion by design engineering, the inspectors questioned why operators had not consequently designated the actions to observe torus trough water level as compensatory measures to maintain operability of secondary containment and whether these actions would now apply to both torus troughs. On March 6, 2010, operators designated these compensatory measures as necessary to maintain secondary containment operability. The inspectors then questioned operators and design engineering about whether the equipment installed to support the compensatory measures should be designated as temporary modifications. EN-DC-136, Revision 5, "Temporary Modifications," Attachment 9.2, states, in part, that "specific temporary physical plant alterations specified for compensatory measures to maintain operability would be a Temporary Modification." On March 25, 2010, system engineering established a corrective action to issue temporary modifications for both torus troughs.

Analysis: The inspectors determined that Entergy's inadequate implementation of their operability determination process (EN-OP-104) and their temporary modification process (EN-DC-136) for compensatory measures that were required to maintain operability of secondary containment was a performance deficiency. Traditional enforcement did not apply, as the issue did not have actual or potential safety consequence, had no willful aspects, nor did it impact the NRC's ability to perform its regulatory function. A review of NRC Inspection Manual Chapter (IMC) 0612, Appendix E, "Minor Examples," revealed that no minor examples were applicable to this finding. The inspectors determined that the finding was more than minor because the finding was associated with the Human

Performance Attribute of the Barrier Integrity Cornerstone, and adversely affected the cornerstone's objective to provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events. Specifically, operations and engineering personnel did not implement operability determination and temporary modification procedures when degraded and/or non-conforming conditions associated with the secondary containment torus troughs were identified. The inspectors determined the significance of the finding using IMC 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." The finding was determined to be of very low safety significance (Green) because the finding only represented an impact to the radiological barrier function provided by secondary containment and the standby gas treatment system.

This finding had a cross-cutting aspect in the area of Human Performance, Work Practices component, because Entergy personnel did not follow procedures. Specifically, Entergy did not implement their operability determination or temporary modification procedures for compensatory measures needed to maintain the secondary containment operable. [H.4(b)]

<u>Enforcement:</u> 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Contrary to the above, Entergy did not accomplish the requirements outlined in procedures for the determination of operability or for the identification of temporary modifications when compensatory measures were identified which were necessary to maintain secondary containment operability. Specifically, between December 30, 2009 and March 6, 2010, Entergy did not adequately implement EN-OP-104 and designate installed remote cameras and level indicating devices as compensatory measures that were required to ensure adequate water level was maintained to keep the secondary containment water seal operable. Additionally, between March 6 and March 25, 2010, Entergy did not adequately implement EN-DC-136 to designate the temporary physical plant alterations (remote cameras and level devices) specified as compensatory measures to maintain operability as Temporary Modifications. Entergy's corrective actions included designating the compensatory measures as necessary to maintain operability for both torus troughs and implementation of temporary modifications for the equipment installed in the plant to support these compensatory measures. The secondary containment torus trough issues are documented in CR-PNP-2009-5295, CR-PNP-2009-5309, and CR-PNP-2010-0014. Because this finding is of very low safety significance and Entergy has entered it into their corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **NCV 05000293/2010002-01, Failure to Implement Operability Determination and Temporary Modification Processes for Compensatory Measures Required to Maintain Operability of Secondary Containment.**

.2 Temporary Modification to Provide 24VDC Power during "A" Battery Testing

a. Inspection Scope (1 sample)

The inspectors reviewed Temporary Modification EC12349, "Provide 24VDC Power during "A" Battery Testing," to determine whether the performance capability of the "A" 24VDC safety related bus had been degraded through the modification. The inspectors reviewed Control Room and procedural drawings, relevant condition reports, and work orders to ensure the temporary modification did not adversely affect the 24VDC system. The inspectors reviewed the annotated drawings to determine whether they properly reflected the temporary modification. The inspectors also walked down the battery and switchgear rooms to ensure tagging was appropriate for the modification.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (7 samples)

The inspectors reviewed seven samples of post-maintenance tests (PMT) during this inspection period. The inspectors reviewed these activities to determine whether the PMT adequately demonstrated that the safety-related function of the equipment was satisfied, given the scope of the work performed, and that operability of the system was restored. In addition, the inspectors evaluated the applicable test acceptance criteria to verify consistency with the associated design and licensing bases, as well as Technical Specification requirements. The inspectors also evaluated whether conditions adverse to quality were entered into the corrective action program for resolution. The documents reviewed during this inspection are listed in the Attachment. The following maintenance activities and their post-maintenance tests were evaluated:

- C-19A Electronics are unresponsive;
- Standby Gas Treatment "A" Train Backdraft and Outlet Damper maintenance and testing;
- High Pressure Coolant Injection (HPCI) Electrical maintenance including various breaker refurbishment and testing;
- HPCI Electrical maintenance including HPCI Condensate Pump Motor Brush replacement, HPCI Auxiliary Lube Oil Pump Motor Brush replacement, and HPCI Gland Seal Condenser Blower Pump Brush replacement;
- Residual Heat Removal Pump "B" Relay maintenance;
- HPCI replacement of Rupture Disk, repair of Temperature Control Valve (TCV-2301-230) and additional mechanical maintenance postwork tests; and
- "A" Emergency Diesel Generator Snubber Valve replacement.

b. Findings

Introduction: The NRC identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for Entergy's failure to promptly correct a condition adverse to quality. Specifically, Entergy did not correct leaking snubber injection valves on the "A" Emergency Diesel Generator (EDG) in a timely manner.

Description: On March 12, 2008, Pilgrim's "A" EDG exhibited a fuel oil leak from its 7R cylinder during the monthly surveillance. The fuel leak was determined to be a symptom of a cracking phenomenon of the fuel injector snubber valve. The snubber valve serves to dampen pulsations from the positive displacement injector pump and to act to keep the fuel tube full on the back stroke of the pump. The "A" EDG was removed from service to replace the snubber valve on the 7R cylinder and was subsequently returned to service. A 10 CFR 21 report from Entergy's Palisades plant was written on April 2, 2008 and listed Pilgrim as a plant susceptible to this cracking snubber valve phenomenon. Specifically, the particular material used in these snubber valves was susceptible to material defects from improper through-hardening during the manufacturing process. Entergy then discovered additional cracked snubber valves on its "A" EDG 9R cylinder on June 9, 2008, during a maintenance overhaul window and then on its "A" EDG 6R cylinder during subsequent post work testing on June 14, 2008. CR-PNP-2008-1894 and CR-PNP-2008-1952 were written. After three of the 18 snubber valves were replaced, Entergy conducted an extent of condition review and determined that nine of the remaining 15 snubber valves on the "A" EDG would be susceptible to cracking due to material defects. The susceptible snubber valves on the "B" EDG previously had been replaced during an overhaul in 2009. The apparent cause recommended replacement of these snubber valves before the next scheduled overhaul in the summer of 2010. This activity did not take place.

On March 10, 2010, Entergy discovered the "A" EDG 2L and 3L fuel cylinders leaking from cracked snubber valves. Entergy removed the "A" EDG from service for emergent maintenance and replaced the affected snubber valves. Their action plan is to replace the remaining seven susceptible snubber valves in June 2010, during the next planned "A" EDG overhaul.

Analysis: The performance deficiency was that Entergy did not promptly correct a condition adverse to quality, cracked snubber injection valves on their "A" EDG. The failure to correct this condition in a timely manner (over two years from identifying the first leaking cylinder) resulted in additional unplanned unavailability for the "A" EDG. Traditional enforcement did not apply; as the issue did not have actual safety consequence, had no willful aspects, nor did it impact the NRC's ability to perform its regulatory function. A review of NRC Inspection Manual Chapter (IMC) 0612, Appendix E, "Minor Examples," revealed that no minor examples were applicable to this finding. The inspectors determined that the finding was more than minor because the finding was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone's objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, unplanned maintenance added additional unavailability to the "A" EDG during snubber valve replacement. The inspectors determined the significance of the finding using IMC 0609.04, "Phase 1 –

Initial Screening and Characterization of Findings." The finding was determined to be of very low safety significance (Green) because, although additional "A" EDG unavailability was incurred, the finding did not result in a loss of system safety function of a single train for greater than its Technical Specifications allowed outage time and did not screen as potentially risk significant due to external initiating events.

This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Corrective Action Program component, because Entergy did not take corrective actions in a timely manner. Specifically, Pilgrim did not replace the "A" EDG snubber valves in a prompt manner after repeated fuel leaks from cracked snubber valves over the previous two years. [P.1(d)]

Enforcement: 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires in part that measures shall be established to assure that conditions adverse to quality, such as defective material and equipment are promptly identified and corrected. Contrary to the above, Entergy was not prompt in correcting defective material in their "A" EDG which led to emergent maintenance and additional unplanned unavailability of the "A" EDG while they replaced cracked snubber valves. Entergy's corrective actions include replacing the seven remaining snubber valves on their "A" EDG with those of a material properly hardened and not susceptible to the same mode of cracking. Entergy has captured these failures in their corrective action program as CRs 2008-0852, 2008-1071, 2008-1894, 2008-1952, and 2010-0898. Because this finding is of very low safety significance and Entergy has entered it into their corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **NCV 05000293/2010002-02, Untimely Corrective Actions to Promptly Correct Leaking Snubber Valves on the "A" Emergency Diesel Generator.**

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (6 samples)

The inspectors witnessed six surveillance activities and/or reviewed test data to determine whether the testing adequately demonstrated equipment operational readiness and the ability to perform the intended safety-related functions. The inspectors reviewed selected prerequisites and precautions to determine if they were met and if the tests were performed in accordance with the procedural steps. Additionally, the inspectors evaluated the applicable test acceptance criteria for consistency with associated design bases, licensing bases, and Technical Specification requirements. The inspectors also evaluated whether conditions adverse to quality were entered into the corrective action program for resolution. The following surveillance tests were evaluated:

- "A" Low Pressure Coolant Injection Quarterly Operability Test, In-Service Test (IST);
- Standby Liquid Control Pump Quarterly and Biennial Capacity and Flow Rate Test;
- "B" Low Pressure Coolant Injection Quarterly Pump Test (IST);
- "A" Emergency Diesel Generator Initiation by Loss of Offsite Power Logic;

- Core Spray System Test (IST); and
- Drywell Leak Detection (Reactor Coolant System).

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 drill observation sample)

The inspectors observed an emergency planning drill on February 24, 2010. The inspectors evaluated the emergency response organization performance in the simulator, in the alternate Technical Support Center, and in the Emergency Operations Facility, for a hostile action based scenario which escalated to a General Emergency. The inspectors assessed the implementation of Emergency Action Level classification and notification decisions as well as Protective Action Recommendation development and notifications. The inspectors also assessed whether Pilgrim's critique of the exercise assessed all of the drill's observations and findings.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY (RS)

Cornerstone: Occupational and Public Radiation Safety

2RS05 Radiation Monitoring Instruments (71124.05)

a. Inspection Scope

During the period between February 8 and 12, 2010, the inspectors performed the following activities to verify that Entergy was ensuring the accuracy and operability of radiation monitoring instrumentation. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, relevant Technical Specifications, and Entergy's procedures.

Inspection Planning

- The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) to identify radiation instruments associated with monitoring area radiological conditions including airborne radioactivity, process streams, effluents, material/articles, and workers.
- The inspectors obtained a listing of all survey instrumentation including air samplers, small article monitors (SAMs), personnel contamination monitors (PCMs), and other

monitors used to detect internal contamination. The inspectors reviewed the list to determine if an adequate number and type of instruments are available to support operations.

- The inspectors obtained and reviewed copies of evaluation reports of the radiation monitoring program since the last inspection.
- The inspectors obtained and reviewed copies of procedures used for instrument source checks and calibrations.
- The inspectors reviewed area radiation monitor set point values and basis.
- The inspectors reviewed the effluent monitor set point basis and the calculational methods provided in the offsite dose calculation manual (ODCM).

Walkdowns and Observations

- The inspectors toured the Turbine and Reactor buildings and observed the condition of the Steam Jet Air Ejector monitors, the Main Stack Ventilation monitors, the Reactor Building Ventilation monitors, and the Radioactive Waste Discharge monitor. These monitor configurations aligned with Pilgrim's ODCM descriptions.
- The inspectors checked the calibration due dates and source check stickers for portable survey instruments ready for issue or in the field. The type of instruments checked included RO-2As, RO-20s, Telepoles, and Ludlum 3s.
- The inspectors observed a technician perform instrument source checks during the back shift. The inspectors verified that the instrument source checks included exposures at each high-range scale. The source check observations included RO-2s, RO-2As, RO-20s, Telepoles, and Ludlum 3s.
- The inspectors verified Area Radiation Monitors (ARM) and Continuous Air Monitors (CAM) were appropriately positioned relative to the radiation source(s) they were intended to monitor. The inspectors compared the monitor response with actual area conditions for several ARMs.
- The inspectors observed the daily source checks for PM-7 #600, SAM #308, and Aptec PMW-2 #52. The inspectors verified the source checks were in accordance with the manufacturer's recommendations and Pilgrim's procedures.

Calibration and Testing Program

Process and Effluent Monitors

- The inspectors verified for more than four effluent monitor instruments that channel calibration and functional tests were performed consistent with radiological effluent technical specifications. The inspectors also verified that the source calibrations use National Institute of Standards and Technology (NIST) traceable sources or secondary measuring that has been calibrated to NIST standard. The inspectors verified that the sources used represent the plant nuclide mix.
- The inspectors verified that effluent monitor alarm set points are established as provided in the ODCM and station procedures.
- There were no changes to effluent monitor set-points during this inspection period.

Laboratory Instrumentation

- The inspectors verified that the daily performance checks and calibration data indicate the frequency of calibration is adequate and there is no degradation of instrument performance.

Whole Body Counter

- The inspectors reviewed the methods and sources used to perform the Whole Body Counter (WBC) checks prior to daily use. The inspectors verified the checks are appropriate and align with the plant's isotopic mix.
- The inspectors reviewed the WBC calibration reports completed since the last inspection. The inspectors verified the calibration sources and phantoms used were appropriate and representative of the plant source term.

Post Accident Monitoring Instrumentation

- The inspectors reviewed the April 18, 2009 calibration records for the Drywell high-range monitors, RIT-1001-606A and RIT-1001-606B.
- The inspectors verified that an electronic calibration for the Drywell high-range monitors was performed and included each decade above 10 rem/hour. The inspectors also verified that a source calibration was performed and included an exposure for at least one decade below 10 rem/hour.
- The inspectors verified the acceptance criteria were reasonable.
- The inspectors reviewed the calibration records and availability for the Main Stack Ventilation and Reactor Building Ventilation high range monitors.
- The inspectors reviewed Pilgrim's capability to collect high-range, post accident iodine effluent samples.
- There were no opportunities to observe electronic or source calibrations of the high range monitors during this inspection.

PMs, PCMs, and SAMs

- The inspectors verified that the alarm set point values for PM 7s, SAM s, and PMW-2 are reasonable to ensure licensed material is not released from Pilgrim.
- The inspectors reviewed the calibration records for PM 7 # 600, SAM # 308, and PMW-2 # 52.

Portable Survey Instruments, ARMs, Electronic Dosimetry, and Air Samplers/CAMs

- The inspectors reviewed calibration records for ARMs, an AMS-4, an RO-20, an RO-2, an RO-2A, a Radeco, a PM-7, a Radiation Air Sampler (RAS) Flow Gauge, a Ludlum 3, and a Ludlum 177. The inspectors reviewed the detector measurement geometry and calibration methods for ARMs and portable radiation survey instruments. The inspectors had a technician demonstrate the use of the instrument calibrator.
- There were no opportunities to review the corrective actions taken for instruments found significantly out of calibration during this inspection.

Instrument Calibrator

- The inspectors reviewed the current output tables for Entergy's portable survey and ARM instrument calibrator unit. The inspectors verified that Entergy periodically measures the calibrator output over the range of the instruments.
- The inspectors verified the calibrator is sent for periodic calibration to a facility that uses NIST traceable sources.

Calibration and Check Sources

- The inspectors reviewed Entergy's 10 CFR 61 source term to verify that the calibration sources used are representative of the types and energies of radiation encountered in the plant.

Problem Identification and Resolution

- The inspectors reviewed thirteen (13) condition reports related to radiation monitoring instrumentation and verified that appropriate corrective actions have been taken or initiated. The inspectors verified that problems are being identified at the appropriate threshold and are properly addressed for resolution.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator (PI) Verification (71151)

Cornerstones: Mitigating Systems and Barrier Integrity

a. Inspection Scope (3 samples)

The inspectors reviewed PI data to determine the accuracy and completeness of the reported data. The review was accomplished by comparing reported PI data to confirmatory plant records and data available in plant logs, Licensee Event Reports (LER), Condition Reports (CRs), and NRC inspection reports. The acceptance criteria used for the review was Nuclear Energy Institute (NEI) 99-02, Revision 6, "Regulatory Assessment Performance Indicator Guidelines" and NUREG-1022, Revision 2, "Event Report Guidelines 10CFR 50.72 and 50.73." The documents reviewed during the inspection are listed in the Attachment. The following performance indicators were reviewed:

- Mitigating System Cornerstone, Safety System Functional Failures from the first quarter of 2009 through the fourth quarter of 2009;

- Barrier Integrity Cornerstone, Reactor Coolant System (RCS) Activity from the first quarter of 2009 through the fourth quarter of 2009; and
- Barrier Integrity Cornerstone, RCS Unidentified Leakage from the first quarter of 2009 through the fourth quarter of 2009.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Items Entered into the Corrective Action Program (CAP)

a. Inspection Scope

The inspectors performed a screening of each item entered into Entergy's CAP. This review was accomplished by reviewing printouts of each Condition Report (CR), attending daily screening meetings and/or accessing Entergy'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 Annual Sample: Review of an Automatic Scram Resulting from a Switchyard Breaker Fault During a Severe Winter Storm, and Momentary Loss of all 345 kV Off-Site Power to the Startup Transformer from a Switchyard Breaker Fault

a. Inspection Scope (1 sample)

The inspectors selected condition reports (CR) PNP-2008-03962 and PNP-2008-03980 as problem identification and resolution (PI&R) samples for a detailed follow-up review. CR PNP-2008-03962 documented an automatic scram due to a switchyard fault during a severe winter storm on December 19, 2008. CR PNP-2008-03980 documented the momentary loss of all 345 kV off-site power to the startup transformer (SUT), X4, while the plant was in hot shutdown on December 20, 2008. Entergy determined that the cause of the automatic scram on December 19, 2008, was conductive snow/ice buildup on the non-conductive porcelain surfaces of the ACB-105 circuit breaker bushing during a severe snow storm. The snow/ice accumulation on the circuit breaker "A" phase bushing resulted in an electrical fault causing a reactor scram with the plant operating at 100% power. Entergy determined that the cause of the momentary loss of all 345 kV off-site power to the SUT was a phase "B" to ground fault on the switchyard line 355 bus section. A directional ground overcurrent relay (DGOR) at the Auburn Street Station facility was incorrectly set and an incorrect signal was sent to trip a breaker (ACB-103) for the 342 off-site power line in the Pilgrim Nuclear Power Station (PNPS) switchyard. This event was initiated by an electrical fault caused by accumulated snow falling from the overhead line 355 bus section and bridging the gap to the "B" phase arc horn. This caused the momentary loss of the 355 line and the 342 line. By design, the 342 line should not have tripped. The 342 line tripped due to an incorrect overcurrent setting of the DGOR at the Auburn Street Station facility. The DGOR was incorrectly set because of an error in a grid computer model used by the Auburn Street Station facility owner to determine the proper setting for the DGOR. As a result, proper clearing of PNPS switchyard faults in the 345 kV switchyard for the 342 line did not occur. The proper setting of the DGOR would have prevented the 342 line from tripping thus maintaining an off-site power source to PNPS.

The inspectors assessed Entergy's problem identification threshold, cause analyses, extent of condition reviews, operability determinations, and the prioritization and timeliness of corrective actions to determine whether Entergy was appropriately identifying, characterizing, and correcting problems associated with these issues and whether the planned or completed corrective actions were appropriate to prevent recurrence. Additionally, the inspectors performed walkdowns of the PNPS 345 kV switchyard to assess if abnormal conditions existed. The inspectors also interviewed cognizant plant personnel regarding the identified issues and implemented corrective actions. Specific documents reviewed are listed in the attachment to this report.

b. Findings and Observations

No findings of significance were identified.

The inspectors determined that Entergy properly implemented their corrective action process regarding the initial discovery of the reviewed issues. The CR packages were complete and included root cause evaluations (RCE), operability determinations, extent of condition reviews, use of operating experience, and corrective actions. Additionally, the elements of the condition reports and RCEs were detailed and thorough. Corrective actions appeared appropriate to minimize the potential of flashover faults in the 345 kV switchyard and prevent recurrence of the momentary loss of both 345 kV off-site power lines. The inspectors determined that corrective actions for the December 19, 2008, event included revising severe weather procedures to provide enhanced monitoring of the PNPS 345 kV switchyard/components during severe snow weather events and included implementation of specific operator actions as a result of degrading conditions (snow/ice buildup) in the switchyard. The corrective actions for the December 20, 2008, event included revising the set-point for the Auburn Street Station facility DGOR for the 342 off-site power line to operate at the appropriate fault current setting. The new setting will allow the 342 offsite power line to remain energized for a fault on the 355 off-site power line, thus maintaining one off-site electrical power source. PNPS coordinated with the interconnection transmission owner, the Auburn Street Station facility owner, and the transmission operator to verify and validate the new fault current setting for the Auburn Street Station DGOR. Additionally, it appeared that Entergy took appropriate corrective actions and post-trip reviews to evaluate and replace damaged switchyard components prior to placing them back into service following the reviewed events.

.3 Annual Sample: Review of Reactor Core Isolation Cooling (RCIC) Discharge and Suction Pressurization After Shutdown From Routine Testing

a. Inspection Scope (1 Sample)

The inspectors reviewed the circumstances leading to the pressurization of the RCIC discharge and suction piping after a January 6, 2010, routine surveillance testing and the actions taken by Entergy to evaluate this condition. Specifically, following quarterly surveillance testing, after the pump was secured, pump discharge pressure slowly rose from the normal pressure of 33 psig to at or near the suction relief valve setpoint of 100 psig over approximately 8 hours. To relieve the pressure, operators opened the pump minimum flow valve from the control room three times, venting the pressure to the torus. Following the third venting the suction pressure did not increase.

The safety concern was the potential for back leakage of the significantly higher temperature (approximately 365 °F) feedwater causing vapor voids in the RCIC pump discharge and suction piping, which could cause water hammer during pump start or damage to the pump, respectively following a pump start. Continuous leakage of feedwater, driven by an approximate 1200 psig, past the RCIC injection check valve, injection double disc gate valve and the CST suction check valve could increase the temperature of the water in the discharge and suction piping above the saturation temperature for the static pressure conditions.

b. Findings/Observations

No findings of significance were identified.

The inspectors reviewed the following and found Entergy's actions adequate to address the condition:

- Condition Report CR-PNP-2010-00063, Received RCIC pump Suction Pressure High alarm, dated January 6, 2010;
- Process Applicability Determination, dated January 12, 2010, used to justify revision 14 to the RCIC high suction pressure alarm response procedure (ARP-C904L-A3), to specify opening the pump minimum flow valve to reduce the pressure;
- Operability/Functionality Evaluation, dated January 19, 2010, which determined that RCIC was operable and that either a small amount of leakage past the injection check valve and normally closed injection double disc gate valve or thermal expansion of initially cold water from the CST caused the pressurization. This included verification, using a work order on January 7, 2010, that the system piping up to the normally closed MOV-49 was full of water using ultrasonic measurement and verification that the piping was at ambient temperature of approximately 80 °F. The speculation was that the downstream disc of MOV-49, which was closed in this test, had been pushed just off its closed seat by the pump's discharge pressure, allowing a small amount of leakage which pressurized the suction side of the pump; and,
- Troubleshooting was conducted during the next quarterly RCIC surveillance test on March 3, 2010, which indicated that the cause was leakage past MOV-49 versus

thermal expansion of colder CST water. Specifically this test closed MOV-48 upstream of MOV-49 following the ST and monitored the pressure between the two valves and the pump suction pressure. Pressure between MOV-49 and MOV-48 increased slowly to just over 190 psig in about three and a half hours; while the pump suction pressure did not change. This indicated that leakage past MOV-49 was the cause of the pressurization.

Based on the above, the inspectors concluded that given the documented condition post-surveillance test leakage past MOV-49 was relatively small and not an operability concern relative to discharge or suction piping voiding. The inspectors also concluded that venting the piping, as directed by the alarm response procedure for high RCIC suction pressure, using the minimum flow valve was an acceptable contingency action.

4OA3 Event Follow-up (71153)

.1 Operator Performance During Condenser Backwash

a. Inspection Scope (1 sample)

The inspectors observed an infrequently performed evolution on March 10, 2010. Specifically, the inspectors observed a plant downpower to support backwashing of the condenser. The inspectors observed the operators reduce power from 100 percent to 46 percent by lowering recirculation flow and inserting control rods. The inspectors reviewed procedural guidance and the power maneuver plan, and observed control room conduct and control of the evolutions. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Loss of Standby Gas Treatment due to Demister Door Open

a. Inspection Scope (1 sample)

On March 25, 2010, an Entergy security officer during rounds discovered an open demister door to charcoal vaults on the "B" Standby Gas Treatment (SBGT) System. The security officer notified the control room and operations declared both trains of SBGT inoperable. Operators entered Technical Specification 3.7.B, which requires Standby Gas to be restored within a 36 hour timeframe. Operations dispatched an operator, closed the demister door, exited the Technical Specification, and made notifications for the loss of SBGT. The inspectors reviewed control room logs, Technical Specifications, and notification requirements.

b. Findings

See Section 4OA7.

.3 <u>(Closed) Licensee Event Report (LER 05000293/2009-001-00), Target Rock Relief Valves Test Pressure Exceeded Limit Due to Setpoint Variance</u>

a. <u>Inspection Scope</u> (1 sample)

On June 15, 2009, Entergy identified that three out of four target rock relief valve pilot assemblies exceeded their Technical Specification pressure limits during routine testing post Refueling Outage 17. Entergy had replaced all four of their pilot assemblies for their main steam relief valves during Refueling Outage 17. NRC review of upward pressure setpoint drift is documented in Regulatory Issue Summary 2000-12, Resolution of Generic Issue 165, "Spring Actuated Safety and Relief Valve Reliability." Additionally, specific safety relief valve issues at Pilgrim are likewise documented in IR 05000293-2007-06, and in the problem identification and resolution section of IR 05000293-2008-005. No new findings or violations of significance were identified during the inspector's review. The LER provided an accurate description of planned follow-up actions related to Pilgrim's safety relief valves. This LER is closed.

b. <u>Findings</u>

No findings of significance were identified.

.4 <u>(Closed) Licensee Event Report (LER 05000293/2009-002-00), Failure to Meet Technical Specification Requirements for Secondary Containment</u>

a. <u>Inspection Scope</u> (1 sample)

The inspectors reviewed Entergy's actions associated with LER 05000293/2009-002-00, which are addressed in the CAP as CR-PNP-2009-5295 and CR-PNP-2009-5309. The event was discussed in NRC Inspection Report (IR)05000293/2009005 and related inspection findings are discussed in Sections 1R18 and 4OA7 of this report. The documents reviewed during the inspection are listed in the Attachment. This LER is closed.

b. <u>Findings</u>

See Sections 1R18 and 4OA7.

4OA6 <u>Meetings, Including Exit</u>

On February 11, 2010, the inspectors performed a Radiation Safety exit meeting with the plant at 2:00 P.M. Kevin Bronson, Site Vice President, attended the meeting. The inspectors verified that no proprietary information was provided to the inspectors during the inspection.

On March 18, 2010, the inspectors presented a debrief of the inspection results to Mr. Stephen Beneduci, Engineering Supervisor, and Mr. Jeffrey Keene, Systems Engineer. This inspection report feeder does not contain proprietary information.

On April 21, 2010, the resident inspectors conducted an exit meeting and presented the preliminary inspection results to Mr. Kevin Bronson, and other members of the Pilgrim staff. The inspectors confirmed that proprietary information provided or examined during the inspection was controlled and/or returned to Entergy and the content of this report includes no proprietary information.

4OA7 Licensee-Identified Violations

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

- Technical Specification (TS) 5.4.1 requires written procedures shall be established, implemented, and maintained covering procedures specified in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Contrary to this, on March 25, 2010, a Demister Door on the "B" train of the Standby Gas Treatment (SBGT), required to be closed following the surveillance activity in procedure 8.M.3-18, was found to be left open by an Entergy security officer conducting normal rounds. SBGT was declared inoperable and then was restored to service. This event is documented in Entergy's corrective action program as CR-PNP-2010-1079. The finding is of very low safety significance because the finding only represents a degradation of the radiological barrier function provided for the SBGT system.

- Technical Specification (TS) 3.7.C.1 requires secondary containment to be operable in the Run, Startup and Hot Shutdown Modes, during movement of recently irradiated fuel assemblies in the Secondary Containment and during operations with a potential for draining the reactor vessel. Contrary to the above, on December 22, 2009, the "A" torus trough, which is required to be maintained at a water level above Reactor Building Close Cooling Water drain pipe openings, was found dry. Secondary Containment was declared inoperable, the torus trough water level was restored and TS 3.7.C.1 was exited. This event is documented in Entergy's Corrective Action Program as CR-PNP-2009-5295 and CR-PNP-2009-5309. The finding was determined to be of very low safety significance (Green) because the finding only represented an impact to the radiological barrier function provided by secondary containment and the standby gas treatment system.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy personnel:

S. Beneduci	Engineering Supervisor
S. Bethay	Director, Nuclear Safety Assurance
K. Bronson	Site Vice President
B. Byrne	Licensing Engineer
S. Das	Electrical Design Engineer
R. Hargat	Radiation Protection Technician
G. Jennings	Radiation Protection Technician
K. Kampschneider	Senior Systems Engineer
J. Keene	Systems Engineer
W. Lobo	Licensing Engineer
J. Martin	Electrical Maintenance Superintendent
M. McDonnell	Operations Assistant Manager
T. McElhinney	Chemistry Manager
D. Noyes	Operations Manager
J. Priest	Radiation Protection Manager
K. Sejkora	Senior Chemist
R. Smith	General Manager Pilgrim Operations
M. Thornhill	Radiation Protection Supervisor

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

NCV 05000293/2010002-01	Failure to Implement Operability Determination Process and Temporary Modification Process for Compensatory Measures Required to Maintain Operability of Secondary Containment.
NCV 05000293/2010002-02	Inadequate Corrective Actions to Promptly Correct Leaking Snubber Valves on the "A" Emergency Diesel Generator.

Closed

LER 05000293/2009-001-000	Target Rock Relief Valves Test Pressure Exceeded Limit Due to Setpoint Variance
LER 05000293/2009-002-000	Failure to Meet Technical Specification Requirements for Secondary Containment

LIST OF DOCUMENTS REVIEWED

Section 1R01

Procedure 5.2.2, Revision 31, High Winds (Hurricane)
Procedure 2.1.37, Revision 25, Coastal Storm – Preparations and Actions
Procedure 2.1.42, Revision 9, Operation during Severe Weather

Section 1R04

Procedure 2.2.24, Revision 46, Standby Liquid Control System
Procedure 2.2.20, Revision 71, Core Spray
UFSAR Volume 2, Section 6; Core Standby Cooling Systems
Equipment Out of Service (EOOS) Tool
Training Manual, RHR System, Revision 1
Open Work Order List for System 10, RHR System
Drawing M241, Sheet 1 & 2, Revision E2, Residual Heat Removal System
Procedure 2.2.21, Revision 76, High Pressure Coolant Injection System
Maintenance Work Schedule the week of March 8, 2010
Procedure 2.2.8, Revision 95, Standby AC Power System (Diesel Generators)

Section 1R05

Fire Hazards Analysis
Fire Zone Data Sheets
PNPS Appendix R Exemption Summary
Exemption Request #11
Procedure 5.5.2, Revision 44, Special Fire Procedure
Procedure 2.2.29, Revision 27, Smoke and Heat Detection Systems
CR-PNP-2010-0325, Lights are out on 91' of the Reactor Building
CR-PNP-2010-00661, Fire Hazard Analysis CRD QUAD combustible loading differs from limits described in Exemption Request #11
Engineering Evaluation #59, Revision 1, Pipe Penetrations with steel plates
Herculite Material Safety Data Sheet
EN-DC-161, Revision 3, Control of Combustibles

Section 1R06

Pilgrim Probabilistic Safety Assessment, Revision 1, App. E, Internal Flood Analysis Section 4.2.1.30, Rupture in the RHR "B" Quadrant (Flood Zone RB-170)
UFSAR Section 10.13.3.3 Radioactive Floor Drainage System
Calculation S&SA 60, Revision 0, Flooding due to ECCS Leakage Outside Containment
P&ID M437, Revision E3, Radwaste Drainage System
Emergency Operating Procedure (EOP)-04, Secondary Containment Control
CR-PNP-2010-632, "B" Quad Housekeeping Issues Identified by NRC Inspector

Section 1R07

Generic Letter 89-13, Service Water Problems Affecting Safety-Related Equipment
Procedure 8.5.3.14.1, Revision 4, Reactor Building Closed Cooling Water (RBCCW) heat exchanger thermal performance test
Calculation M710, Revision 0, Heat Exchanger Performance Testing
Boston Edison's response to NRC Generic Letter 89-13, Service Water Problems Affecting Safety-Related Equipment
Procedure 8.5.3.14.2, Revision 2, RHR Heat Exchanger Thermal Performance Test
FSAR Chapter 10.5, Reactor Building Closed Cooling Water System
FSAR Chapter 10.7, Salt Service Water System
Specification M591, Salt Service Water & RBCCW Safety-related Piping and Heat Exchanger Inspection Maintenance and Test Requirements in response to Generic Letter 89-13
CR-PNP-2010-659, Observed difference between "A" and "B" RHR Hx Fouling Factors and lack of instrument uncertainty calibrations
CR-PNP-2010-660, EDG heat exchangers not included in Heat Sink Performance Monitoring Program
CR-PNP-2010-00739, NRC Resident annual heat exchanger inspection, identified deficiencies to be corrected under CR-PNP-2010-00659

Section 1R11

Combined Functional Drill 10-01 Scenario
Simulator Operator Aids 13, 24 and 40
Simulator Procedure 5.3.21 Excerpt
Simulator Procedure 5.3.23 Excerpt
EP-IP-100.1, Revision 4, Emergency Action Levels (EALs)
Combined Functional Drill 10-01 Emergency Planning Performance Indicator Submittals

Section 1R12

Procedure 8.B.21, Revision 39, Emergency Lighting Units
CR-PNP-2010-0310, ELBU-131R would not illuminate
CR-PNP-2010-0311, ELBU-151R would not illuminate
CR-PNP-2010-0312, ELBU-113R would not illuminate
CR-PNP-2010-0313, ELBU-118R would not illuminate
E-Lights System Health Report
Maintenance Rule Basis Document, Revision 1, Emergency Lighting System
CR-PNP-2009-5295, Secondary Containment Inoperable due to Loss of Torus Trough Water Level
CR-PNP-2009-5295, Functional Failure Evaluation for Torus Trough
CR-PNP-2010-0469, Incorrect Maintenance Rule Functional Failure Determination for CR-PNP-2009-5295
Maintenance Rule Basis Documents
EN-DC-204, Revision 2, Maintenance Rule Scope and Basis
Maintenance Rule Basis Document, Revision 2, Reactor Building Closed Cooling Water System
RBCCW System Health Report

CR-PNP-2010-0130, Broken Bolt on RBCCW suction Header and Effectiveness Review
CR-PNP-2010-0834, Inadequate Functional Failure Determination for RBCCW

Section 1R13

Control Room Logs
Daily Risk Sheets for Week of January 4, 2010
CR-PNP-2010-0047, Incorrect Risk Assessment during Reactor Core Isolation Cooling (RCIC) Testing
Risk Profile Sheets for January 27, 2010
Drawing M203, Sheet 1, Revision 48, PI&D Main Steam System
Control Logs for January 15, 2010
CR-PNP-2010-00223, PT Fuse Failure, Relay 160-609
Equipment Out of Service Quantitative Risk Assessment Tool
Procedure 1.5.22, Revision 12, Risk Assessment Process
CR-PNP-2010-220, Plant evaluation of the impact of external events may not be consistent with NRC guidance
CR-PNP-2010-130, ½ inch bolt to support clamp for RBCCW broken
Control Room Logs for March 9, 2010
Daily Risk Sheet for March 9, 2010

Section 1R15

CR-PNP-2010-0223, Received PT Fuse
Failure for "B" EDG Relay 160-609 and Operability Evaluation
Control Room Logs for January 15, 2010
Electrical Drawing E-5-152-5BC, Revision E2, Relay Schematic for EDG "B"
CR-PNP-2010-0229, RBM "A" Received Rod Block due to loss of input signal and operability evaluation
CR-PNP-2009-2932, RBM "A" Received Rod Block on June 25, 2009
UFSAR Section 7.7.4.3, Rod Block Interlocks
Technical Specifications
Alarm Response Procedure C3LC-E8, Revision 11, B-18 MCC Enclosure Temperature High
CR-PNP-2010-0572, B-18 MCC Temperature is high and associated operability evaluation
Final Safety Analysis Report (FSAR) Section 8.4, Auxiliary Power Distribution System
Control Room Logs for February 12, 2010
EN-OP-104, Revision 3, Operability Determinations
CR-PNP-2009-5309, Small amount of water found at the base of the "A" Torus trough
CR-PNP-2010-14, Basis for CR-PNP-2009-5295 does not appear to encompass existing conditions
CR-PNP-2009-5295, Low water level in Torus trough "A" results in secondary containment inoperability
UFSAR Chapter 10.7.6, Safety Evaluation of RBCCW Flooding Pipe directed into Torus troughs
Operations Compensatory Measures List
Calculation No. C15.0.3381, Allowable Secondary Containment Leakage Area
Secondary Containment Relative Pressure Data for December 22, 2009
UFSAR Chapter 5.3, Secondary Containment System
Technical Specifications 3/4.7.C and Bases, Secondary Containment

EN-OP-104, Revision 4, Operability Determinations
CR-PNP-2010-614, GE Modeling Error in ATWS Analysis
CR-PNP-2010-247, ATWS Suppression Pool Temperature Limit Higher than Previously Analyzed
Procedure 2.2.22 Reactor Core isolation Cooling System, Revision 77
Procedure 8.5.5.1, RCIC Pump Quarterly and Biennial Operability Flow rate and Valve Test at Approximately 1000 PSIG, Revision 72, completed January 6, 2010.
Procedure 2.2.22.5, RCIC Injection and Pressure Control, Revision 14
ARP-C8R, Alarm Response Procedure, Revision 6

Section 1R18

EN-DC-136, Revision 5, Temporary Modifications
CR-PNP-2009-5309, "A" Torus Trough Leakage
CR-PNP-2009-5295, Torus Room troughs not filled with water
UFSAR Section 8.7, 24 Volt DC Power System
Training Manual Schematics for 24VDC System
WO 51565583, Task 1, 24VDC Battery Acceptance Test
Procedure 3.M.3-36.8, Revision 1, Temporary Power for 24VDC Bus "A" or "B"
Temporary Modification Logs
CR-PNP-2009-3064, Temperature Mod Control Low Level Trend
EC12349, Revision 0, Provide Temporary Modification to Support 24VDC "A" Battery Testing

Section 1R19

WO 00222150, C-19A (West) Unresponsive to Control Panel
Post Work Test C19 Functional Check Data Sheet from Chemistry Procedure 7.4.17
CR-PNP-2010-0147, C19A Electronics are locked up
WO 52028609, Task 1, SBGT Exhaust Fan VEX-210A Backdraft Damper Maintenance
Post Work Test – Inservice leakage check on VEX-210A, Task 3 on WO 52028609
WO 52190076, Task 1, SGBT "A" Outlet Damper, MO-N-109, Maintenance
Post Work Test on MO-N-109, Task 2 on WO 52190076
CR-PNP-2010-0452, Leaking noted on discharge side of VEX-210A and operability evaluation associated with CR
CR-PNP-2010-0233, Standby Gas Treatment "B" train trouble alarms on flow
CR-PNP-2010-0587, Standby Gas Discharge High Rad Alarm
WO 51569467 01, Breaker Preventive Maintenance on 72-971, DC to MO-2301-10
WO 51654818 05, Breaker Testing 72-944, DC to MO-2301-3
Procedure 8.Q.3-4, Revision 51, 125/250V DC Motor Control Center and Breaker Panel Testing and Maintenance
WO 51569467 03, Breaker PM on 72-971, Pre-testing
WO 51569467 04, Breaker PM on 72-971, Pre-test Overload Relay
WO 51569467 02, 72-971 Post Work Testing
Procedure 8.I.32, Revision 6, Determination of Limiting Stroke Time Acceptance Criteria for In-Service Testing and Appendix B Test Programs Power Operated Valves
WO 51551667 01, Breaker Bucket PM for DC to MO-2301-36 (72-831)
WO 51551667 04, 72-831 Pre-test Overload Relay
WO 51551667 03, Pre-test Breaker 72-831
WO 51551667 02, Post Maintenance Test Breaker 72-831

WO 52217942 01, Replace HPCI Gland Seal Hotwell Pump Motor Brushes (P220)
Procedure 3.M.3-7.3, Revision 7, Generic Brush Inspection and Maintenance
WO 52037059 01, Insulation Test of P220
WO 52217942 02, Post Work Test P220 Brush Replacement
WO 52199276 01, Replace HPCI Auxiliary Lube Oil Pump Motor Brushes (P229)
WO 52217943 01, Replace P223 Motor Brushes
WO 52037058 01, Insulation Test of P223
WO 52038461 01, "B" RHR Pump Relay Testing
WO 52213597 01, "B" RHR Pump (P203B) Motor Preventive Maintenance
Procedure 8.Q.3-2, Revision 21, RHR/Core Spray Pump Motor Preventive Maintenance
Procedure 3.M.3-1, Revision 126, A5/A6 Buses 4KV Protective Relay Calibration/Functional Test
WO 00209562 01, "D" RHR Pump (P-203D) Insulation Test
WO 52038819 01, "D" RHR Pump Relay Calibration/Functional Test
CR-PNP-2010-00902, During NRC review of work package, discrepancies/enhancements to both procedures and work orders were identified
WO 00208897, Repair Insulation from HPCI Turbine
WO 51533345, Tasks 2 and 3, Repair Oil Leaks on TCV-2301-230
CR-PNP-2008-2647, Oil Leaks above TCV-2301-230
WO 00216914, Task 1, HPCI booster pump oil change
Procedure 3.M.4-17.4, Revision 34, Lubrication sampling and change procedure
WO 00174770, Clean out HPCI steam supply drain line strainer YS-8048
WO 51692743, Task 1, HPCI Stop valve balance chamber
Procedure 3.M.3-61.5, Revision 39, Emergency Diesel Generator Two-Year Overhaul Preventive Maintenance
WO 00229139 01, Replace Fuel Injector Snubbers on "A" EDG X-107A (2L/3L)
EN-MA-102, Revision 4, Attachment 9.2, Inspection Report
WO 00229139 02, Replace Fuel Injector Snubbers on "A" EDG X-107A (2L/3L), Post Work Test
Procedure 8.9.1, Revision 114, Emergency Diesel Generator and Associated Emergency Bus Surveillance
EN-MA-101, Revision 9, Fundamentals of Maintenance
EN-WM-107, Revision 2, Post Maintenance Testing

<u>Section 1R22</u>

Briefing Checklists
Procedure 8.5.2.2.1, Revision 51, LPCI System Loop "A" Operability
Procedure 8.4.1, Revision 70, Standby Liquid Control Pump Quarterly and Biennial Capacity and Flow Rate Test
CR-PNP-2010-191, PNPS 8.4.1 aborted due to inability to set test conditions
WO 52223812 01, 8.5.2.2.2 (Section 8.1) P-LPCI System Loop "B" PP Valve Quarterly Operability
Procedure 8.5.2.2.2, Revision 42, LPCI System Loop "B" Operability – Pump Quarterly and Biennial (comprehensive) Flow Rate Tests and Valve Tests
Technical Specifications
Procedure 8.M.2-2.10.8.5, Revision 44, Diesel Generator "A" Initiation by Loss of Offsite Power Logic
CR-PNP-2010-0885, Limiting Condition of Operation is Incorrect in Surveillance Procedure
CR-PNP-2010-0881, Relay 127A-504/1 and 127A-504/2 were out of spec.
Procedure 8.5.1.1, Revision 54, Core Spray System Operability – Pump Quarterly and Biennial

Comprehensive Flow Rate Tests and Valve Tests
TS 4.5, Core and Containment Cooling Systems
RCS Leakage Data Sheets for CY 2009
Performance Indicator Process Data Sheets from Procedure EN-LI-114, Revision 4, for first quarter of 2009 through fourth quarter of 2009
Control Room Logs

Section 1EP6

Combined Functional Drill 10-01 Scenario
Simulator Operator Aids 13, 24 and 40
Simulator Procedure 5.3.21 Excerpt
Simulator Procedure 5.3.23 Excerpt
EP-IP-100.1, Revision 4, Emergency Action Levels (EALs)
Combined Functional Drill 10-01 Emergency Planning Performance Indicator Submittals

Section 2RS05

Procedures:

6.5-160	Calibration of the Area Radiation Monitoring System, Revision 34
6.5-170	Calibration of Ventilation System Radiation Monitors using ARM Type Sensor/Converters, Revision 23
6.6-116	Source Calibration of the Containment High Radiation Monitoring System Using the MDH 2025 X-Ray Monitor, Revision 13
6.7.2-101	Calibration of the Fastscan Whole Body Counter, Revision 6
7.4.12	Calibration of the SJAE Offgas Process Radiation Monitors, Revision 24
7.4.14	Calibration of Main Steam Line Process Radiation Monitors, Revision 27
7.4.24	RBCCW Process Radiation Monitors, Revision 38
7.4.29	Source Calibration of General Atomic High-Range Noble Gas Monitors, Revision 21
7.4.42	Calibration of the NUMAC Gaseous PRMs, Revision 25
7.4.47	Calibration of the Radwaste Effluent PRM, Revision 12
7.4.63	Process Radiation Monitor Setpoints, Revision 10
7.10.3	PRM Calibration Check, Revision 21
7.10.8	Main Stack, Reactor Building Vent, and Radwaste PRM Functional Check and Source Check, Revision 24
3.M.2-6.1	RBCCW Process Radiation Monitor Calibration, Revision 21
3.M.2-6.4	NUMAC Process Radiation Monitor Calibration, Revision 17
3.M.2-19	High Range Effluent Monitor Calibration, Revision 22
8.E.8	Offgas Instrumentation Calibration, Revision 47
8.M.1-13	Main Steam Line High Radiation Calibration and Functional Test, Revision 50
8.M.2-4.1	Air Ejector Offgas Log Radiation Monitor Calibration, Revision 31
8.M.2-4.2	Air Ejector Offgas Radiation Monitor Functional Channels "A" and "B", Revision 34
8.M.2-4.4	High Range Effluent Monitoring Functional Test, Revision 14
8.M.2-4.6	Offgas Post-Treatment Radiation Monitors Functional Test, Revision 10

EN-DC-143	System and Component Health Reports, Revision 9
EN-DC-143-01	System Health Report Supplemental Guidance, Revision 0

Condition Reports:

2009
00127, 01115, 01126, 02200, 02515, 02584, 03142, 03331, 03466, 04707, 05005, 05184, 05186

Calibration Records

Instrument Type	Serial #	Calibration Date
Portable Radiation Survey Instruments		
AMS-4	200	7/13/2009
Ludlum 177	257827	9/3/2009
RO-20	151	8/28/2009
RO-2	3439	9/14/2009
RO-2A	3295	9/14/2009
Contamination Monitors		
PMW-2	52	2/3/2010
SAM-9	308	6/29/2009
PM-7	600	6/10/2009
Air Sampler Flow Gauge		
RAS	R-33	8/31/2009
H809V	4397	9/2/2009

Source Calibration Form

Serial #	Isotope	Date
N-273	Cs-137	7/15/2009
N-562	Cs-137	4/6/2009
N-360	Cs-137	11/6/2009
N-12	Cs-137	11/10/2009
N-321	Cs-137	8/24/2009
N-265	Cs-137	9/25/2009

Area Radiation Monitors (ARM)

Procedure 6.5-160 Revision 34

RIS-1815-2A	Control Room
RIS-1815-2B	TIP Room
RIS-1815-2C	Radwaste Truck Bay
RIS-1815-2D	Reactor Building 23 SE Access Area
RIS-1815-3A	Condensate Pump area
RIS-1815-3B	Radwaste Hallway
RIS-1815-3D	Reactor Building 117 New Fuel Storage

RIS-1815-3E	Reactor Building 117 Floor Plug Area
RIS-1815-3F	Spent Fuel Pool Area
RIS-1815-8C	Radwaste Sump Area
RIS-1815-8D	Chemical Waste Receiver Tank

Process and Effluent Monitors

Monitor	Description	Procedure	Revision
RM-1705-3B	Steam Jet Air Ejector Offgas	7.4.12	23
RM-1705-5B	"B" Augmented Offgas	7.4.42	25
		7.10.3	21
RM-1705-8A	Refuel Floor Exhaust	6.5-170	23
RM-1705-8B	Refuel Floor Exhaust	6.5-170	23
RM-1705-8C	Refuel Floor Exhaust	6.5-170	23
RM-1705-8D	Refuel Floor Exhaust	6.5-170	23
RM-1705-18A	"A" Main Stack	7.4.42	25
		7.10.3	21
RM-1705-18B	"B" Main Stack Ventilation	7.10.3	21
RM-1705-32A	"A" Reactor Building Ventilation	7.10.3	21
RM-1705-32B	"B" Reactor Building Ventilation	7.4.42	25

High Range Monitors

RT-1001-608	Main Stack Ventilation	7.4.29	21
RT-1001-609	Reactor Building Ventilation	7.4.29	21
RT-1001-610	Turbine Building Ventilation	7.4.29	21
RIT-1001-607A	Torus Area	6.1-210	18
RIT-1001-607B	Torus Area	6.1-210	18
RIT-1001-606A	Drywell Area	6.1-210	18
RIT-1001-606B	Drywell Area	6.1-210	18

Flow Rate Instruments

Instrument	Description	Procedure	Revision
FY-3725	Steam Jet Air Ejector	8.E.8	45
FT-8116	Reactor Building Ventilation	8.E.8	45
FT-9368	Main Stack Ventilation	8.E.8	45
8-2264-FT-1	Reactor Building Ventilation	8.E.8	46
8-2247-1-FT-1	Main Stack Ventilation	8.E.8	46

Audits and Self-Assessments

O2C-PNPS-2009-0337
O2C-PNPS-2009-0546

Other

Report # 9646 Radcal Corporation
Report # 7076 Radcal Corporation

Section 40A1

NEI 99-02, Revision 6, Regulatory Assessment Performance Indicator Guideline
LER 2009-001-00, Target Rock Relief Valves' Test Pressure Exceeded Limit Due to Setpoint Variance
LER 2009-002-00, Failure to Meet Technical Specification Requirements for Secondary Containment
LER 2009-002-01, Failure to Meet Technical Specification Requirements for Secondary Containment, Revision 1
EN-LI-114, Revision 4, Performance Indicator Process
EN-LI-114, Revision 4, Attachment 9.2, NRC Performance Indicator Technique/Data Sheet
CR-PNP-2010-00653, LER 2009-002-00, did not include a loss of safety function categorization
Pilgrim Station Coolant Iodine Values for CY-2009
Performance Indicator Process Data Sheets from Procedure EN-LI-114, Revision 4, for first quarter of 2009 through fourth quarter of 2009
RCS Leakage Data Sheets for CY-2009

Section 40A2

Condition Reports

2008-03962
2008-03980
2009-01092
2010-00977*

* CR initiated as a result of this inspection.

Completed Surveillance Procedures

3.M.3-5, Electrical Termination Procedure, Rev. 26, Completed 11/03/08
8.M.3-20, Line 342 and Line 355 Protective Relaying Telecommunications System Testing, Rev. 0, Completed 04/18/09

Drawings

E1, Single Line Diagram, Station, Rev. 21
29050F, 345 kV One Line & Relay Diagram STA650 -- Switchyard, Rev. E1
Isometric Drawing 6498-612-1 RCIC Discharge Piping, Revision EO

Licensing Documents

Pilgrim Nuclear Power Station Technical Specifications
Pilgrim Nuclear Power Station Updated Final Safety Analysis Report

Licensee Event Report

LER 2008-006-00, Automatic Scram Resulting from Switchyard Breaker Fault during Winter Storm
LER 2008-007-00, Momentary Loss of all 345 kV Off-Site Power to the Startup Transformer from Switchyard Breaker Fault

Miscellaneous

Maintenance Rule SSC Basis Document, 345 kV, Main/Unit Aux/Start-up Transformers, Generator Excitation, & ISO-Phase Bus, Rev. 10
Technical Basis for Revising the Maintenance Rule Unavailability Performance Criteria for Lines 342 and 355
Vendor Manual, GE-Hitachi HVB, Inc., 362 kV 40-63 kA – 2000/3000 A, IPO SF_6 Gas Circuit Breaker Two Cycle Interruption, Rev. 5
White Paper, 20 Dec. 08 Switchyard Flashover Event Review with Regard to Restart

Preventive Maintenance Basis Documents

PMBD-284, All Components & Structures Located Inside 650 & Transformers (Including Spares) Outside the Switchyard, Rev. 0
PMBD-285, ACB-102, ACB-103, ACB-104, ACB-105, 345 kV GE-Hitachi SF_6 Circuit Breaker, Rev. 0
PMBD-286, 23 kV Switchyard, Rev. 0
PMBD-288, Coupling Capacitor Voltage Transformers (CCVT), Rev. 0
PMBD-289, 345 kV Disconnect Switches, Rev. 0
PMBD-291, Switchyard Relays – Bulk Power Protection, Rev. 0

Procedures

EN-LI-102, Corrective Action Process, Rev. 14
EN-LI-118, Root Cause Analysis Process, Rev. 12
2.1.37, Coastal Storm-Preparations and Actions, Rev. 25
2.1.42, Operation During Severe Weather, Rev. 9
Procedure 2.2.22 Reactor Core isolation Cooling System, Revision 77
Procedure 8.5.5.1, RCIC Pump Quarterly and Biennial Operability Flow rate and Valve Test at Approximately 1000 PSIG, Revision 72, completed January 6, 2010.
Procedure 2.2.22.5, RCIC Injection and Pressure Control, Revision 14
ARP-C8R, Alarm Response Procedure, Revision 6
EN-OP-104, Operability Determination Process, Revision 4

System Health Reports

23 kV and Transformers, 3rd Quarter 2009
23 kV and Transformers, 4th Quarter 2009
345 kV, 3rd Quarter 2009
345 kV, 4th Quarter 2009

Work Orders

00170309
00176793
00185329
00189064

Section 40A3

Control Room logs for March 10, 2010
Risk Profile for Condenser Backwash on March 10, 2010
Power Maneuver Plan
Procedure 2.1.14, Revision 101, Station Power Changes
Control Room logs for March 25 and March 26, 2010
50.72 Notification for Loss of Standby Gas Treatment System
Technical Specifications
Licensee Event Report 2009-001, Revision 0, Target Rock Relief Valves Test Pressure Exceeded Limit Due to Setpoint Variance
NUREG 1022, Event Reporting Guidelines

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ARM	Area Radiation Monitors
CAM	Continuous Air Monitors
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
DGOR	Directional Ground Over Current Relay
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
HPCI	High Pressure Coolant Injection
HX	Heat Exchanger
IMC	Inspection Manual Chapter
IR	Inspection Report
IST	Inservice Testing
kV	kilovolt
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PCM	Personnel Contamination Monitors

PI	Performance Indicator
PI&R	Problem Identification and Resolution
PMT	Post-Maintenance Test
PNPS	Pilgrim Nuclear Power Station
RBCCW	Reactor Building Closed Cooling Water
RCE	Root Cause Evaluations
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SAM	Small Article Monitors
SBGT	Standby Gas Treatment
SDP	Significance Determination Process
SSC	Structure, System or Component
SUT	Startup Transformer
TCV	Temperature Control Valve
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
VDC	volts direct current
WBC	Whole Body Counter
WO	Work Order