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

February 5, 2009

EA-09-013

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 05000293/2008005 – EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Bronson:

On December 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Pilgrim Nuclear Power Station (PNPS). The enclosed report documents the results, which were discussed on January 7, 2009, with you 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 one NRC-identified finding, and one self-revealing finding of very low safety significance (Green). Both of these findings were determined to involve violations of NRC requirements. 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 (NCV)s, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest the NCVs in this report, 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 Resident Inspector at the Pilgrim Nuclear Power Station.

Additionally, this report closes one issue involving an incorrect entry into Technical Specification (TS) 4.0.3, Surveillance Requirement Applicability, after you determined Reactor Protection System (RPS) time response testing had not been conducted on several RPS scram contactors in 2007 (Unresolved Item (URI) 05000293/2007-003-04). The NRC determined that TS 3.1, Reactor Protective System, should have been entered vice entering TS 4.0.3, because the surveillance on this portion of the RPS system had never been performed. Although the incorrect entry into TS 4.0.3 is a violation of NRC requirements, the NRC identified no performance deficiency and that discretion is warranted because: (1) licensee current basis documents do not specifically clarify the distinction between a missed surveillance and one that has never been performed, (2) the licensee subsequently completed the surveillance testing satisfactorily, and (3) the issue was of very low safety significance. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, and the Region I Regional Administrator, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy, and refrain from issuing enforcement action for the violation.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, 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/ Original Signed By;

David C. Lew, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000293/2008005
w/Attachment: Supplemental Information

cc w/encl:
Vice President, Operations, Entergy Nuclear Operations
Vice President, Oversight, Entergy Nuclear Operations
Senior Manager, Nuclear Safety & Licensing, Entergy Nuclear Operations
Senior Vice President and COO, Entergy Nuclear Operations
Assistant General Counsel, Entergy Nuclear Operations
R. Walker, Director, Radiation Control Program, Commonwealth of Massachusetts
W. Irwin, Chief, CHP, Radiological Health, Vermont Department of Health
The Honorable Therese Murray
The Honorable Vincent deMacedo
Chairman, Plymouth Board of Selectmen
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J. Miller, Senior Issues Manager
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Electric Power Division, Commonwealth of Massachusetts
R. Shadis, New England Coalition Staff
D. Katz, Citizens Awareness Network
W. Meinert, Nuclear Engineer
J. Giarrusso, MEMA, SLO
Commonwealth of Massachusetts, Secretary of Public Safety

documents do not specifically clarify the distinction between a missed surveillance and one that has never been performed, (2) the licensee subsequently completed the surveillance testing satisfactorily, and (3) the issue was of very low safety significance. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, and the Region I Regional Administrator, to exercise enforcement discretion in accordance with Section VII.B.6 of the Enforcement Policy, and refrain from issuing enforcement action for the violation.

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Sincerely,

/RA/ Original Signed By:

David C. Lew, Director
Division of Reactor Projects

Docket No. 50-293
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Enclosure: Inspection Report 05000293/2008005
w/Attachment: Supplemental Information

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REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	05000293/2008005
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Pilgrim Nuclear Power Station (PNPS)
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	October 1, 2008 through December 31, 2008
Inspectors:	M. Schneider, Sr. Resident Inspector, Division of Reactor Projects (DRP) B. Smith, Resident Inspector, DRP R. Fuhrmeister, Senior Project Engineer, DRP R. Rolph, Health Physicist, Division of Reactor Safety (DRS) A. Ziedonis, Reactor Inspector, DRS
Approved By:	David C. Lew, Director Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000293/2008-005; 10/01/2008-12/31/2008; Pilgrim Nuclear Power Station; Maintenance Risk Assessments and Emergent Work Control, and Event Followup

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

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50.65(a)(4) for Entergy’s failure to conduct a risk assessment for emergent maintenance on the High Pressure Coolant Injection (HPCI) system injection valve. Specifically, the failure to conduct a risk assessment resulted in Entergy not recognizing an increase in risk to a Yellow condition, and therefore no risk management actions were taken. Entergy entered this issue into their corrective action program. Corrective actions will include revising attachments in Entergy’s Technical Specification requirements procedure to perform a risk review as a result of emergent maintenance activities.

  This finding was more than minor because Entergy failed to consider the unavailability of a risk significant system where the outcome of the risk assessment would have been a change in a risk management category. The inspectors conducted an evaluation in accordance with IMC 0609, “Significance Determination Process,” Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process.” The finding was determined to be of very low safety significance (Green) because the Incremental Core Damage Probability Deficit for the timeframe that HPCI was removed from service was significantly less than 1E-6. The inspectors determined that this finding had a cross-cutting aspect in the area of Human Performance, Decision Making, because Entergy did not use a systematic process to make a risk-significant decision when faced with an unexpected plant condition. [H.1(a)] (Section 1R13)

- Green. A self-revealing Green non-cited violation (NCV) of TS 5.4.1, “Procedures”, was identified for a procedure which resulted in an inadvertent isolation of the Reactor Core Isolation Cooling (RCIC) system. Specifically, the procedure was previously revised and a step was inadvertently placed out-of-order. The procedure incorrectly instructed technicians to remove relay contact blockers, or “boots”, before clearing an isolation signal which resulted in the system isolation. Entergy entered this issue into their corrective action program. Corrective actions will include revising this procedure and reviewing other surveillance procedures that had been revised at the same time.

This finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone. Isolating the RCIC system affected the cornerstone objective of ensuring the availability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated this finding using IMC 0609.04, "Phase 1 – Initial Screening and Characterization of Findings". This finding was of very low safety significance because it was not a design or qualification deficiency, did not represent a loss of system safety function, did not represent an actual loss of a single train system for greater than the Technical Specification allowed outage time, and was not made risk-significant because of external events. The inspectors determined that this finding had a cross-cutting aspect in the area of Human Performance, Resources, because Entergy did not ensure that the procedure was complete and accurate. [H.2(c)] (Section 4OA3)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station (PNPS) operated at or near 100 percent power during the majority of the inspection period. However, on December 19, 2008, Entergy scrammed from 100 percent power due to a load reject during a winter storm. Entergy resumed 100 percent power operation on December 24, 2008. The plant remained at or near 100 percent 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 Seasonal Susceptibility

a. Inspection Scope (1 sample)

The inspectors reviewed actions taken by the licensee in preparation for the onset of cold weather during the week of November 2, 2008. The inspectors reviewed Procedure 8.C.40, Seasonal Weather Surveillance, and verified that selected steps had been completed. The inspectors walked down selected areas addressed in the procedure to determine if heat tracing as well as plant heating systems were properly working. The inspectors also walked down exterior portions of the Condensate Storage Tanks and the Station Blackout Diesel. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Impending Storm

a. Inspection Scope (1 sample)

On December 19, 2008, a significant winter storm was tracking to impact the Pilgrim plant that afternoon and into the evening. The inspectors reviewed Entergy's preparations for the impending snow storm as well as for the high winds expected to accompany the storm. The inspectors reviewed Entergy's severe weather procedures including coastal storm preparations and operations during severe weather (specifically, snow storm preparations). The inspectors also reviewed the stated plant risk given the external risk increase and compared this to equipment that was out of service to determine if there was an overall increase in risk. The inspectors conducted 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 or if there were any vulnerabilities to snow accumulation (such as emergency diesel generator ventilation), and thereby impact to safety related equipment. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Partial System Walkdowns (71111.04Q)

a. Inspection Scope (5 samples)

The inspectors performed five partial system walkdowns during this inspection period. The inspectors reviewed the documents listed in the Attachment to determine the correct system alignment. The inspectors conducted 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 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:

- Instrument and Service Air Systems During Maintenance on K-111 Air Compressor;
- Automatic Depressurization System with Reactor Core Isolation Cooling unavailable due to maintenance;
- “B” Spent Fuel Pool System while in standby;
- “A” train Salt Service Water (SSW) System with “D” SSW unavailable; and
- “B” Core Spray System with High Pressure Coolant Injection out for maintenance.

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 required 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, fire barriers, and any related compensatory measures. 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 the inspection are listed in the Attachment. The fire protection areas reviewed were:

- Reactor Building/El.23'-0" up to El.51'-0", East Side – Fire Area 1.9, Fire Zone 1.9;
- "A" Switchgear and Load Center Room – Fire Area 1.9, Fire Zone 2.2;
- "B" Switchgear and Load Center Room – Fire Area 1.10, Fire Zone 2.1;
- Spent Fuel Pool Cooling Pumps and Heat Exchanger Area – Fire Area 1.9, Fire Zone 1.13; and
- Vital Motor Generator Set Room – Fire Area 1.9, Fire Zone 3.5.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Licensee-Administered Annual Operating Tests

a. Inspection Scope (1 sample)

On November 17, 2008, a region-based inspector conducted an in-office review of results of the licensee-administered annual operating tests and comprehensive written exams for 2008. 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." The inspector verified that:

- Crew failure rate was less than 20 percent. (Crew failure rate was 0 percent)

- Individual failure rate on the dynamic simulator test was less than or equal to 20 percent. (Individual failure rate was 0 percent)

- Individual failure rate on the walk-through test was less than or equal to 20 percent. (Individual failure rate was 0 percent)

- Individual failure rate on the comprehensive written exam was less than or equal to 20 percent. (Individual failure rate was 0 percent)

- Overall pass rate among individuals for all portions of the exam was greater than or equal to 75 percent. (Overall pass rate was 100 percent)

b. Findings

No findings of significance were identified.

.2 Licensed Operator Training

a. Inspection Scope (1 sample)

The inspectors observed licensed operator training on November 18, 2008. Specifically, the inspectors observed classroom Senior Reactor Operator (SRO) training on the

Severe Accident Guidelines (SAGs), Core Mitigation Strategies, and the Emergency Operating Procedures (EOP). The lectures discussed SAG and EOP entry conditions, roles and responsibilities for the SROs, and the phenomenology of severe accidents. The inspectors assessed the training to determine if the training adequately prepared the SROs to determine what actions to take in a severe accident situation and when to enter the SAGs. The inspectors reviewed the lesson plans and applicable training objectives to determine if they had been achieved. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q)

.1 Review of Functional Failures

a. Inspection Scope (3 samples)

The inspectors reviewed three functional failure determinations conducted in accordance with Entergy procedures and the requirements of 10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. The inspectors reviewed the system maintenance rule functions, the basis for the conclusion that the issues were considered functional failures, and the potential for common cause and extent of condition. The inspectors also reviewed data to verify whether or not the functional failures resulted in placing the systems in (a)(1). The inspectors reviewed system health reports to determine if actions taken were reasonable and appropriate. In addition, the inspectors reviewed Entergy's condition reports and corrective actions. The documents reviewed during the inspection are listed in the Attachment. The functional failure determinations reviewed were:

- CR-PNP-2008-02120, Post Accident Sample System has inadequate heat tracing;
- CR-PNP-2008-02469, Standby Gas Treatment System root valve does not fully shut; and
- CR-PNP-2008-03338, High Pressure Coolant Injection Valve relay in circuit breaker cabinet fails.

b. Findings

No findings of significance were identified.

.2 Review of K-117 Air Compressor (a)(1) Action Plan

a. Inspection Scope (1 sample)

The inspectors reviewed the (a)(1) corrective action plan for the K-117 air compressor unavailability exceeding the (a)(2) unavailability criteria for items such as: (1) appropriate work practices; (2) identifying and addressing common cause failures; (3) scoping in accordance with 10 CFR 50.65(b) of the maintenance rule (MR); (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(a)(1) or (a)(2); and (8) appropriateness of performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSCs/functions classified as (a)(1). The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope (3 samples)

The inspectors evaluated three online 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 verified the appropriate use of Entergy's risk assessment tool, Equipment Out of Service (EOOS), and entry into appropriate risk categories. 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 documents reviewed during the inspection are listed in the Attachment. The inspectors reviewed the conduct and adequacy of maintenance risk assessments for the following maintenance and testing activities:

- Yellow Risk, During Reactor Core Isolation Cooling and Air Compressor K-111 Maintenance and Testing Activities;
- Emergent Risk of Inoperability of the High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems; and
- Forced Outage Shutdown Risk Assessments.

b. Findings

Introduction. The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50.65(a)(4) for Entergy's failure to conduct a risk assessment for emergent maintenance on the High Pressure Coolant Injection (HPCI) system injection valve. Specifically, the failure to conduct a risk assessment resulted in Entergy not recognizing an increase in risk to a Yellow condition, and therefore no risk management actions were taken.

Description. At 7:44 p.m. on October 21, 2008, operators received an alarm in the control room and determined the cause was a loss of control power for a HPCI injection valve. Entergy declared HPCI inoperable and entered the Limiting Condition for Operation (LCO) for Technical Specification (TS) 3.5.C.2, "HPCI System." This LCO requires that the system be made operable within 14 days, otherwise be in cold shutdown within 24 hours.

With HPCI unavailable, risk management actions are generally taken to protect those systems that provide redundancy for its function, such as the Reactor Core Isolation

Cooling system. Entergy did not conduct a risk assessment nor recognize the plant risk condition was “Yellow” and therefore did not take any risk management actions. Entergy restored HPCI operability and exited the LCO eight hours later.

Analysis. The performance deficiency associated with this finding is that Entergy did not perform a review of the increased risk while HPCI was inoperable and, as a result, did not take risk management actions as required by 10 CFR 50.65(a)(4). This finding is associated with the human performance attribute of the Mitigating Systems cornerstone and is more than minor because Entergy failed to consider the unavailability of a risk significant system where the outcome of the risk assessment would have been a change in risk management category. The inspectors conducted an evaluation in accordance with IMC 0609, “Significance Determination Process,” Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process.” The finding was determined to be of very low safety significance (Green) because the Incremental Core Damage Probability Deficit for the timeframe that HPCI was removed from service was significantly less than 1E-6.

The inspectors determined that this finding had a cross-cutting aspect in the area of Human Performance, Decision Making, because Entergy did not use a systematic process to make a risk-significant decision when faced with an unexpected plant condition. [H.1(a)]

Enforcement. 10 CFR 50.65(a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” states, in part, that “...the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities.” Contrary to the above, from October 21, 2008 to October 22, 2008, Entergy failed to assess the increased risk that resulted from HPCI unavailability. As a result, Entergy did not recognize a “Yellow” risk condition and did not take any risk management actions. Corrective actions will include revising attachments in Entergy’s Technical Specification requirements procedure to perform a risk review as a result of emergent maintenance activities. Because this violation was of very low safety significance (Green) and was entered into the licensee’s corrective action program (CR-PNP-2008-03792), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000293/2008005-01, Failure to Conduct a Risk Assessment for Emergent Maintenance on the High Pressure Coolant Injection System)**

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (5 samples)

The inspectors reviewed five operability determinations associated with degraded or non-conforming conditions to determine if the operability determination was justified and if the mitigating systems or those affecting barrier integrity 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 licensee performance against related Technical Specification and UFSAR requirements. The documents reviewed during the inspection are listed in the Attachment. The inspectors reviewed the following degraded or non-conforming conditions:

- CR-PNP-2008-03049, Air Void in High Pressure Coolant Injection (HPCI) Suction Line;
- CR-PNP-2008-03015, HPCI Cabling In-Service Aging Needs to be reviewed;
- CR-PNP-2008-03404, Shutdown Transformer Breaker (A802) won't close remotely from the Control Room;
- CR-PNP-2007-04801, Restriction Orifices Missing from Recirculation Pump Instrument Lines; and
- CR-PNP-2008-03611, Thermography identifies a hot spot in Air Cooled Breaker (ACB) 104.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Permanent Modification

a. Inspection Scope (1 sample)

The inspectors reviewed Permanent Modification ERO2115031, "Change Orientation of PSV-8008", and the associated 10 CFR 50.59 screening, to determine whether the licensing bases and performance capability of the associated system had been degraded through the modification. A walkdown of the "A" Residual Heat Removal Heat Exchanger was performed to determine if the PSV-8008 valve's new orientation would be subject to additional stress or be impacted by other adverse conditions. The inspectors reviewed system drawings to determine whether they reflected the permanent modification. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Temporary Modification

a. Inspection Scope (1 sample)

The inspectors reviewed Temporary Modification Engineering Change (EC) 7768, "Temp. Mod. Required to Document Heat Detection System Disabled on Pre-Action Sprinkler System for the Turbine Bearings", to determine whether the performance capability of the Fire Protection System had been degraded through the modification. The inspectors reviewed the Updated Fire Hazards Analysis, procedures, and the 10 CFR 50.59 screening to ensure the temporary modification did not adversely affect fire protection program attributes. The inspectors reviewed control room drawings to determine whether they properly reflected the temporary modification. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (6 samples)

The inspectors reviewed six 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 TS requirements. The inspectors also evaluated whether conditions adverse to quality were entered into the corrective action program for resolution. The documents reviewed during the inspection are listed in the Attachment. The following maintenance activities and their post-maintenance tests were evaluated:

- Reactor Core Isolation Cooling Outboard Isolation Valve Electrical Maintenance;
- Open and Inspect Residual Heat Removal (RHR) Pump Discharge Check Valve 1001-67A;
- Replace Undervoltage Relay for High Pressure Coolant Injection Pump Injection Valve MO-2301-08;
- "A" RHR Motor Operated Valve Preventive Maintenance and Breaker, Diagnostic, and Relay Testing for MO-1001-23A, MO-1001-16A, MO-1001-7A, MO-1001-34A, MO-1001-18A, MO-1001-37A, MO-1001-36A and RHR Pump "A" Relays;
- Overhaul of Salt Service Water Pump 208D; and
- Replace Current Transformers on the F-15 Substation Feeding the Shutdown Transformer.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope (1 sample)

The inspectors reviewed the outage plan and shutdown risk assessments for a forced, non-refueling outage conducted from December 19, 2008, through December 23, 2008. The outage was conducted following a plant transient due to a load reject and subsequent reactor plant scram. The load reject was the result of a significant fault in a switchyard breaker during a severe winter storm. During this outage, the inspectors observed plant shutdown activities including the outage activities listed below. The documents reviewed during the inspection are listed in the Attachment.

- Hot Shutdown Control
- Shutdown Risk Assessment and Risk Management
- Implementation of TS
- Outage Control Center Activities
- Plant Startup
- Licensee identification and resolution of problems identified during and related to outage activities

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (2 samples)

The inspectors reviewed two samples of surveillance activities 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 TS requirements. The inspectors also evaluated whether conditions adverse to quality were entered into the corrective action program for resolution. The documents reviewed during the inspection are listed in the Attachment. The following surveillance tests were evaluated:

- Standby Liquid Control In-Service Testing (IST); and
- "B" Emergency Diesel Generator Operability Testing.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (1 sample)

During the period of October 20 through 23, 2008, the inspectors conducted the following activities to verify that the licensee was properly implementing physical, administrative, and engineering controls for access to locked high radiation areas, and other radiologically controlled areas (RCA) during power operations. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, relevant Technical Specifications, and the licensee's procedures. This inspection activity represents the completion of one (1) sample relative to this inspection area.

Plant Walkdown and Radiation Work Permits (RWP) Reviews

The inspectors examined Pilgrim's physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within the spent fuel pool. The inspectors toured the spent fuel pool area and reviewed the procedure for handling highly radioactive objects. The inspectors also toured the Traversing Incore Probe (TIP) Room and Torus Room areas. The inspectors observed the postings and barricades in each area and reviewed surveys and RWPs for the areas including the electronic personal dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy.

b. Findings

No findings of significance were identified.

2OS2 As Low As Reasonably Achievable (ALARA) Planning and Controls (71121.02)

a. Inspection Scope (5 samples)

During the period October 20 through 23, 2008, the inspector conducted the following activities to verify that the licensee was properly implementing operational, engineering, and administrative controls to maintain personnel exposure ALARA during routine plant operation. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and the licensee's procedures. This represents the completion of five samples relative to this inspection area.

Inspection Planning

The inspectors requested a list of the work activities ranked by actual exposure that were completed during refueling outage (RFO) 16. The inspectors reviewed the RWP and ALARA documentation for the five highest dose jobs for RFO16. The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors reviewed the exposure estimates and compared the estimates with the actual dose received.

Radiation Worker Performance

The inspectors observed radiation worker performance prior to entering the RCA. The inspectors questioned workers relative to their individual and department dose goals and their understanding of the previous day's goals and the actual dose received.

Declared Pregnant Workers

The inspectors requested exposure results and monitoring controls employed for declared pregnant workers with respect to the requirements of 10 CFR 20. There were no declared pregnant workers since January 1, 2008.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

a. Inspection Scope (1 sample)

During the period of October 20 through 23, 2008, the inspectors conducted the following activities to evaluate the adequacy of the licensee's program to maintain Self Contained Breathing Apparatus (SCBA). This inspection activity represents the completion of one sample relative to this inspection area.

Self-Contained Breathing Apparatus (SCBA) Maintenance and User Training

The inspectors reviewed the status and surveillance records of SCBA staged and ready for use in the plant. The inspectors observed the inspection of SCBA in the control room and compared the records with the actual equipment staged for use. The inspectors verified that control room operators and other emergency response personnel are trained.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator (PI) Verification (71151)

.1 Mitigating Systems

a. Inspection Scope (2 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, CRs, System Health Reports, and NRC inspection reports. The acceptance criteria used for the review was Nuclear Energy Institute (NEI) 99-02, Revision 5, "Regulatory Assessment Performance Indicator Guidelines." Documents reviewed during the inspection are listed in the Attachment. The following performance indicators were reviewed:

- Emergency Diesel Generators (EDG) from the fourth quarter 2007, through the third quarter of 2008; and
- Cooling Water (Salt Service Water/RBCCW) from the fourth quarter 2007, through the third quarter of 2008.

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope (1 sample)

The inspectors reviewed implementation of the licensee's Occupational Exposure Control Effectiveness Performance Indicator (PI) Program. Specifically, the inspector reviewed recent ACTION reports, and associated documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures against the criteria specified in Nuclear Energy Institute (NEI) 99-02, Revision 5, "Regulatory

Assessment Performance Indicator Guidelines," to verify that all occurrences that met the NEI criteria were identified and reported as performance indicators. This inspection activity represents the completion of one sample relative to this inspection area; completing the annual inspection requirement.

b. Findings

No findings of significance were identified.

.3 RETS/ODCM Radiological Effluent Occurrences

a. Inspection Scope (1 sample)

The inspectors reviewed relevant effluent release reports for the period of January 1, 2007, through December 31, 2007, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences that exceed 1.5 millirem / quarter whole body or 5.0 millirem / quarter organ dose for liquid effluents; 5 millirads / quarter gamma air dose, 10 millirads / quarter beta air dose, and 7.5 millirads / quarter for organ dose for gaseous effluents. This inspection activity represents the completion of one sample relative to this inspection area; completing the annual inspection requirement.

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 the licensee's CAP. This review was accomplished by reviewing printouts of each CR, 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 Annual Sample: Operator Workarounds

a. Inspection Scope (1 sample)

The inspectors performed the annual review of operator workarounds to verify Entergy was identifying operator workaround problems at an appropriate threshold and entering them into the corrective action program. The inspectors reviewed identified workarounds to determine whether the mitigating system function was affected, whether the operator's ability to implement abnormal and emergency operating procedures was affected, and whether appropriate procedures had been updated to reflect actual plant conditions. The inspection was accomplished through personnel interviews, plant tours, and review of station documents. The documents reviewed during the inspection are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified. Operator workarounds have been identified and entered into the corrective action program for resolution. No unrecognized impacts to operator or system performance were identified, and corrective actions have been implemented to restore the affected systems.

.3 Annual Sample: Safety Relief Valve Leakage

a. Inspection Scope (1 sample)

The inspectors selected the issue of Safety Relief Valve (SRV) leakage as an inspection sample for in-depth review because of recent forced plant shutdowns due to SRV leakage. Additionally, SRV leakage has been a long-standing issue at Pilgrim Nuclear Power Station (PNPS). This inspection was conducted to determine if Entergy was taking appropriate corrective actions to address SRV leakage.

The inspectors reviewed procedures, condition reports, engineering evaluations, root cause analyses, and interviewed plant personnel to assess Entergy's problem identification, evaluation, and corrective action effectiveness with respect to SRV leakage. Additionally, the inspectors reviewed the Technical Specifications and Updated Final Safety Analysis Report to assess the adverse impact of SRV leakage with respect to design basis requirements. The documents reviewed during this inspection are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified.

Pilgrim Nuclear Power Station (PNPS) Safety Relief Valves are of the two-stage Target Rock-type design, consisting of a pilot-stage assembly and a main-stage assembly. Industry Operating Experience has shown that two-stage Target Rock SRVs exhibit some amount of pilot-stage leakage during plant operation. Additionally, industry operating experience has quantified SRV leakage in terms of SRV tailpipe temperature, as well as an upward setpoint drift impact. PNPS Technical Specifications require that an engineering evaluation be performed to justify continued operation with elevated tailpipe temperatures. On a number of such occasions, Entergy has implemented an Operational Decision Making Issue (ODMI) to establish administrative limits for the maximum allowable SRV tailpipe temperature for continued plant operation. The operating limits were established to maintain SRV setpoint drift within the ±1% tolerance required by Technical Specifications.

SRV pilot-stage leakage has challenged PNPS throughout the plant's operating history. Entergy has been forced to shutdown the plant from full power operation on three occasions. On each occasion, SRV tailpipe temperatures approached the administrative limits imposed in the respective ODMIs for March 2004, December 2007, and April 2008. Following the March 2004 shutdown, the leaking valves were removed from service and a root cause analysis was performed as a result of high, as-found setpoint testing results. The root cause was determined to be corrosion bonding of the pilot valve disc/seat, with a contributing cause of insulation deficiencies. The contributing cause determination stated that proper fitting SRV insulation is critical for Target Rock SRVs, to reduce the propensity of leakage and eliminate the conditions conducive to corrosion binding. Corrective actions included refurbishing the pilot valve discs, as well as SRV insulation enhancements. The December 2007 shutdown also identified inadequate SRV fitting insulation as one of the root causes for the SRV leakage (CR-PNP-2007-04936). The inspector noted Entergy has self-identified deficiencies with SRV insulation dating back to 1993.

The December 2007, and April 2008, shutdowns identified inadequate simmer margin as one of the root causes (CR-PNP-2007-04936) for SRV pilot-stage leakage. Simmer margin is defined as the pressure difference between SRV setpoint and plant normal operating pressure. At PNPS, the plant simmer margin is limited by a ±1% Technical Specification tolerance (i.e., margin) for the allowable SRV setpoint. PNPS transient and accident analyses have shown this tolerance is required to maintain peak reactor vessel pressures within the code allowable limits. The inspector noted Entergy has self-identified insufficient simmer margin in a number of condition reports dating back to 2004. Additionally, General Electric has issued Service Information Letters to highlight the relationship between pilot-stage leakage and plant operating simmer margin. At the time of this inspection, Pilgrim was investigating corrective action options to address simmer margin, as documented in CR-PNP-2007-04936. The options being pursued included contracting with vendors to increase plant simmer margin via analysis and/or modifications, and completing an independent root cause analysis of pilot valve disc/seat leakage. Entergy was also pursuing the option for development of a new pilot valve disc/seat design.

At PNPS, the effects of SRV upward setpoint drift, even due to small amounts of leakage, are magnified by the limited SRV setpoint tolerance allowed by TS due to the designed plant relief capacity. The inspector found Entergy's planned corrective actions, to address the contributing elements of SRV leakage, to be appropriate.

.4 Annual Sample: Review of Aggregate Impact of Significant Events Which Had Occurred During 2007

a. Inspection Scope (1 sample)

The inspectors selected CR-PNP-2007-04865 for detailed review. The CR was written in December 2007, to evaluate whether there were any additional insights or trends to be identified from an aggregate review of significant events which had occurred during 2007. The inspectors reviewed the licensee's analysis and recommendations for corrective actions. The documents reviewed during the inspection are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified. Entergy reviewed significant events which occurred during 2007, and categorized the events as to their root and contributing causes, whether any regulatory findings had been issued, their impact on plant operation, and the aggregate impact on plant performance (i.e., equipment or human performance). Entergy determined that several of the issues related to the performance of the Emergency Diesel Generators (EDG) and conducted a separate analysis of these events. TI-176 has been conducted to evaluate current EDG testing, the results of which are documented in Section 4OA5. Entergy also reviewed the remaining events and determined that, in some cases, expanding the scope of the evaluation of a given event or condition to review other systems or programs would have been appropriate. As a result, Entergy has instituted a corrective action for the Operations Department to include, as part of their review of degraded condition operability, the need to consider other systems or components that may have a similar vulnerability to the degraded condition. In addition, Entergy will be conducting training with Engineering Department staff to similarly consider other systems, programs, or components when evaluating degraded conditions for extent of condition and corrective actions.

The inspectors reviewed Entergy's corrective actions and evaluated more recent issues to determine if Entergy applied the principles discussed above. The inspectors noted instances where both Operations Department and Engineering Department staffs evaluated a given issue for its applicability beyond the condition itself and, as a result, specified additional corrective actions. The inspectors concluded that these initiatives should improve Entergy's effectiveness in evaluating and correcting issues and in identifying other areas for improvement that may not have been previously considered.

.5 Semi-Annual Review to Identify Trends

a. Inspection Scope (1 sample)

The inspectors performed a review of Entergy's Corrective Action Program (CAP) and associated documents to identify trends that could indicate the existence of a more significant safety issue. The review was focused on repetitive equipment and corrective maintenance issues, but also considered the results of daily inspector CAP item screening. The review included issues documented in CAP trend reports and the site CAP performance indicator data. The review focused on the six month period of July 2008, through December

2008, although the inspectors also evaluated previous trend results for CRs from June 2007, through June 2008, which were discussed in NRC Inspection Reports 05000293/2008003 and 05000293/2007005. Documents reviewed during the inspection are listed in the attachment.

b. Findings and Observations

No findings of significance were identified. In NRC Inspection Report 05000293/2008003, the inspectors concluded that corrective actions to improve configuration control at Pilgrim had not been in effect long enough to conclude whether they were effective. As a result, the configuration control low level trend originally initiated in the fourth quarter report of 2007, continued to be monitored during the past two quarters. The inspectors noted that Entergy's corrective actions appear to have been effective in reducing the number of mispositioning errors. Corrective actions have included conducting operator fundamental training on precise plant control and human performance, generating lessons learned, and generating a standard for mispositioned components and a fleet procedure on configuration control.

In addition to the corrective actions discussed in NRC Inspection Report 05000293/2008003, the Operations department instituted a Mentorship Pilot Program to augment additional reactor operators to assist non-licensed operators in their professional development. Entergy has focused their program in the areas of operator fundamentals, human performance, shift turnover, preparation for shift activities, and operator engagement during briefings. Corrective actions undertaken by the Operations Department appear to have improved the performance of configuration control and, as a result, this low level trend is considered closed. No additional low level trends were identified which would indicate the presence of a broader safety issue.

4OA3 Event Follow-up (71153)

.1 Unplanned Reactor Core Isolation Cooling (RCIC) Isolation

a. Inspection Scope (1 sample)

On the evening of October 6, 2008, operators inadvertently isolated the RCIC system while performing Surveillance Procedure 8.M.2-2.6.3, RCIC Steam Line High Temperature Instrument Functional Test. Specifically, “boots” which were installed to block applicable relays were removed before auto-isolation trip signals had cleared. The RCIC system isolated due to a Group 5 signal which closed the inboard and outboard steam supply valves, which resulted in the actuation of the turbine trip throttle valve. The Group 5 isolation was reset and the operators placed the RCIC system back in its standby lineup.

The inspectors reviewed operator logs, applicable procedural requirements, and technical specifications. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

See Section 4OA3.5.

.2 Entergy Response to a Fire in the Health Physics Calibration Laboratory

a. Inspection Scope (1 sample)

On October 29, 2008, a fire occurred in the Health Physics (HP) instrumentation calibration laboratory located in the Operations and Maintenance (O&M) building. Following the start of the electric fire pump and the generation of O&M building smoke alarms, the on-site Fire Brigade identified that a fire had occurred in the HP calibration lab and that the fire suppression system had actuated and extinguished the fire. Operators requested off-site assistance from the Plymouth Fire Department and both organizations entered the room to verify that the fire suppression system had extinguished the fire. In addition, Entergy declared an Unusual Event due to the occurrence of a fire in the protected area for which off-site assistance was requested. The inspectors responded to the site to evaluate Entergy's actions in response to the fire and to assess any impact on the licensed radiological materials located in the HP calibration laboratory. The inspectors reviewed Entergy's immediate actions to respond to the fire, root cause investigation, event timeline, and corrective actions. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.3 Operator Response to Unplanned Reactor Core Isolation Cooling (RCIC) Unavailability

a. Inspection Scope (1 sample)

On October 22, 2008, operators determined that the RCIC flow controller was inoperable due to aged power supply capacitors. The industry recommended service life is 7 to 10 years and the RCIC capacitors had been in service from 21 to 30 years. Operations declared RCIC inoperable and entered TS 3.5.D, "Reactor Core Isolation Cooling (RCIC) System", a 14-day shutdown action statement. The control panel flow controller was replaced with the flow controller from the alternate shutdown panel (ASP) and operations entered TS 3.12, "Fire Protection" due to the inoperable ASP. During post-installation testing, the flow controller from the ASP was not able to maintain rated flow at the required pressure. A refurbished controller intended to replace the one from the ASP was then installed in the control panel. This flow controller was able to achieve and maintain rated pressure and flow. Operations then declared the system operable and exited TS 3.5.D. Entergy initiated a 10 CFR 50.72, 8-hour, non-emergency notification report for a condition that could have prevented the fulfillment of the safety function of a mitigating system. Entergy conducted subsequent bench testing of the aged capacitors from November 6, 2008 to November 8, 2008 and determined that these capacitors would have been able to perform their function in the required mission time. As a result, Entergy retracted their 10 CFR 50.72 report on December 9, 2008. The inspectors reviewed Technical Specifications, control room logs, interviewed operations and engineering personnel, and reviewed the basis for the retraction of the 10 CFR 50.72 report. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.4 Unplanned High Pressure Coolant Injection (HPCI) Isolation

a. Inspection Scope (1 sample)

On the afternoon of November 20, 2008, operators inadvertently isolated the HPCI system while performing surveillance procedure 8.M.2-2.5.3, Attachment 1, HPCI High Steam Line Temperature. Specifically, in-series, high temperature relays were both inadvertently opened causing a Group 4 isolation signal. A Group 4 isolation signal closes both the inboard and outboard steam supply valves, which rendered the HPCI system inoperable. The Group 4 isolation was reset and the operators placed the HPCI system back in its standby lineup using Procedure 2.2.21, High Pressure Coolant Injection. The inspectors reviewed operator logs, applicable procedural requirements, and Technical Specifications. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.5 Reactor Scram on Load Reject from failure of 345 KV Switchyard Circuit Breakers 104 and 105

a. Inspection Scope (1 sample)

On December 19, 2008, operators responded to a reactor scram from 100% reactor power due to a load reject from the main generator. During a coastal winter storm, an electrical fault developed on the transformer side of circuit breakers 104 and 105 (flashover of the ACB-105 "A" phase generator bushing resulted in a significant current to ground fault) resulting in the main transformer differential relay, main transformer overcurrent relay, and main transformer distance relay actuations to open breakers 104 and 105. Without circuit breakers 104 and 105 to route power to the electrical grid, the turbine, and hence, the reactor scrammed due to a main turbine trip initiated by a main generator load reject. As a result of the reactor scram and load reject, three of four safety relief valves opened briefly, which is expected for the condition. Entergy also received a Group Two isolation signal for Secondary Containment and a Group Six isolation signal for Reactor Water Cleanup. Both of these isolation signals were expected for the situation. However, Y3 and Y4, two of Pilgrim’s 120 VAC safety related instrument buses, remained de-energized following the trip, an unexpected response. Y3 and Y4 are 120 VAC electrical buses which power mitigating system instrumentation including containment isolation logic, SRV acoustic monitoring, and other safety related equipment. This issue was investigated and corrected prior to startup. In addition, a 10 CFR 50.72 notification was generated due to the valid actuation of the Reactor Protection System. The inspectors responded to the control room, reviewed reactor plant parameters and operator response to this event. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.6 Loss of Start-Up Transformer Power Supply from 345KV Switchyard Circuit Breakers ACB 102 and 103

a. Inspection Scope (1 sample)

On December 20, 2008, operators responded to a momentary loss of the 345 KV power supply from the ACB 102 and 103 switchyard circuit breakers. This resulted in the loss of buses A1 through A6 and their associated loads. The emergency diesel generators responded as designed to provide power to vital buses A5 and A6. Group Isolations 1, 2 and 6 were generated on the Reactor Protection Systems signal (Main Steam Isolation Valves (MSIV) closed, primary sample valves isolated, Reactor Water Clean Up isolated and the Reactor Building Ventilation system isolated). The Reactor Core Isolation Cooling and High Pressure Coolant Injection Systems were operated in manual to maintain reactor vessel level and pressure. ACB 102 and 103 re-closed and Buses A1 through A4 were recovered and Buses A5 and A6 were restored to the Start-Up Transformer power supply. MSIV's were opened and Group Isolation Signals were reset. The licensee generated a 10 CFR 50.72 8-hour report to the Nuclear Regulatory Commission discussing this event. The inspectors responded to the control room, reviewed reactor plant parameters and operator response to this event. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.7 (Closed) LER 05000293/2008-003-00, Reactor Core Isolation Cooling (RCIC) System Declared Inoperable During Surveillance Testing due to Procedure Error

a. Inspection Scope (1 sample)

The inspectors evaluated LER 05000293/2008-003-00, "RCIC System Declared Inoperable during Surveillance Testing Due to Procedure Error". This LER is closed with a finding.

b. Findings

Introduction. A self-revealing Green non-cited violation (NCV) of TS 5.4.1, "Procedures", was identified for a procedure error which resulted in an inadvertent isolation of the

Reactor Core Isolation Cooling (RCIC) system. Specifically, the procedure was previously revised and a step was inadvertently placed out-of-order which resulted in the isolation of RCIC.

Description. On the evening of October 6, 2008, Entergy inadvertently isolated the RCIC system while performing Surveillance Procedure 8.M.2-2.6.3, “RCIC Steam Line High Temperature Instrument Functional Test”. When performing this surveillance, Instrumentation and Controls (I&C) technicians use “boots” to block relay contacts and prevent closure of the RCIC steam supply line isolation valves. The I&C technicians followed the procedure, which incorrectly instructed them to remove the “boots” before clearing the auto-isolation trip signals that would close the valves. This resulted in a Group 5 isolation signal that isolated the RCIC system by closing the inboard and outboard steam supply valves, actuating the RCIC turbine trip throttle valve, and rendering the RCIC system inoperable. The operators reset the Group 5 isolation and placed the RCIC system back in its standby lineup within one hour.

The licensee had previously revised several surveillance procedures for the RCIC system and the High Pressure Coolant Injection (HPCI) System. During the revision to Procedure 8.M.2-2.6.3, the step to remove the “boots” was placed before the step to clear the auto-isolation signal. Corrective actions will include revising this procedure and reviewing other procedures that had been revised at the same time.

Analysis. The performance deficiency associated with this finding was that Entergy introduced an error into a procedure during revision that resulted in an inadvertent isolation of the RCIC system. This finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone. Isolating the RCIC system affected the cornerstone objective to ensure the availability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated this finding using IMC 0609.04, “Phase 1 – Initial Screening and Characterization of Findings”. This finding was of very low safety significance because it was not a design or qualification deficiency, did not represent a loss of system safety function, did not represent an actual loss of a single train system for greater than the TS allowed outage time, and was not made risk-significant because of external events.

The inspectors determined that this finding had a cross-cutting aspect in the area of Human Performance, Resources, because Entergy did not ensure that the procedure was complete and accurate. [H.2(c)]

Enforcement. Technical Specification 5.4.1, “Procedures”, requires that written procedures be maintained as recommended in NRC Regulatory Guide (RG) 1.33, “Quality Assurance Program Requirements,” Revision 2, Appendix A, February 1978. RG 1.33, Appendix A, Section 8 includes procedures for surveillance tests. Contrary to this, Procedure 8.M.2-2.6.3 was not adequately maintained, because it included an error that resulted in an isolation of the RCIC system. Corrective actions will include revising this procedure and reviewing other surveillance procedures that had been revised at the same time. Because this finding is of very low safety significance and Entergy has entered it into their corrective action program (CR-PNP-2008-03182), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000293/2008005-02, Procedural Error Resulting in Unplanned RCIC Isolation)**

4OA5 Other Activities

1. (Closed) URI 05000293/2007003-04 Application of TS 4.0.3 When It Was Discovered That a Surveillance Had Never Been Performed

On June 25, 2007, Entergy informed the Nuclear Regulatory Commission (NRC) staff that it had missed a Technical Specification (TS) surveillance requirement to perform time response testing of four Reactor Protection System (RPS) scram contactors. During their review, Entergy identified that the four RPS scram contactors had never been tested. Entergy evaluated the operability of the RPS system and determined that the system remained operable and that TS 4.0.3, "Surveillance Requirement Applicability," would allow a delay period up to the limit of the specified surveillance frequency. The inspectors questioned Entergy regarding the applicability of TS 4.0.3 given that the time response test had never been performed on the RPS scram contactors, as compared to missing a surveillance test following satisfactory initial system baseline testing that originally showed system operability. As a result of this implementation of TS 4.0.3, Entergy failed to take action in accordance with TS 3.1, "Reactor Protective System," which constituted a violation of NRC requirements. Entergy later modified the applicable surveillance procedures and successfully response time tested all RPS scram contactors.

In Task Interface Agreement (TIA) 2008-004, the NRC staff disagreed with Entergy on its implementation of TS 4.0.3 and considered Entergy to have been in violation of TS 3.1, "Reactor Protection System," as a result. Discretion is warranted because: (1) licensee current basis documents do not specifically clarify the distinction between a missed surveillance and one that has never been performed, (2) the licensee subsequently completed the surveillance testing satisfactorily, and (3) the issue was of very low safety significance, since when the correct testing was accomplished, it was completed satisfactorily indicating that the timing of the reactor scram function was not negatively impacted. Accordingly, the NRC staff is exercising enforcement discretion for the TS 3.1 violation in accordance with Section VII.B.6 of the NRC Enforcement Policy and no violation will be issued. **Enforcement Action (EA) 09-013, Failure to Enter TS 3.1 When a Surveillance Requirement Was Not Met. URI 05000293/2007003-04 is closed.**

2. Implementation of Temporary Instruction (TI) 2515/176 – Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing

a. Inspection Scope

The objective of TI 2515/176, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing," was to gather information to assess the adequacy of nuclear power plant emergency diesel generator

(EDG) endurance and margin testing as prescribed in plant-specific technical specifications (TS). The inspectors reviewed emergency diesel generator ratings, design basis event load calculations, surveillance testing requirements, and emergency diesel generator vendor's specifications and gathered information in accordance with TI 2515/176. The inspector assessment and information gathered while completing this TI was discussed with licensee personnel. This information was forwarded on to the Office of Nuclear Reactor Regulation for further review and evaluation.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

On September 11, 2008, the inspectors presented the preliminary inspection results of a Problem Identification and Resolution sample to Mr. B. Sullivan, Nuclear Engineering Director, Mr. S. Bethay, Safety Assessment Director, and other members of the Entergy staff. Following additional in-office review, the inspectors conducted a final exit meeting via teleconference on December 16, 2008, with Mr. S. Bethay and other members of Entergy staff. The inspectors verified that no proprietary information is documented in this report.

On October 23, 2008 at 9:30 A.M., the Occupational Radiation Safety exit meeting was held by phone with Mr. Joe Lynch, Licensing Manager.

On October 31, 2008, an exit meeting of the results of Temporary Instruction (TI) 2515/176 was conducted. The preliminary inspection results were presented to Mr. Stan Wollman, Engineering Supervisor, and other members of the Pilgrim staff. The inspector confirmed that no proprietary information was provided or examined during the inspection.

On January 7, 2009, the resident inspectors conducted an exit meeting and presented the preliminary inspection results to Mr. Kevin Bronson, Site Vice President, and other members of the Pilgrim staff. The inspectors confirmed that no proprietary information was provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:

K. Bronson	Site Vice President
R. Smith	General Manager Pilgrim Operations
S. Bethay	Director, Nuclear Safety Assurance
B. Sullivan	Director, Engineering
W. Cody	ALARA Technician
J. Lamoureux	Sr. Project Manager
W. Lobo	Licensing Engineer
J. Lynch	Licensing Manager
W. Mauro	Supervisor, Radiological Engineering
C. Minott	Sr. Project Manager
D. Noyes	Operations Manager
R. O'Neill	Operations Outage Manager
S. Paul	Operations Supervisor
J. Priest	Radiation Protection Manager
M. Santiago	Supervisor, Operations Training
T. Trainor	Outage Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

09-013	EA	Failure to Enter TS 3.1 When a Surveillance Requirement Was Not Met
05000293/2008-005-01	NCV	Failure to Conduct a Risk Assessment for Emergent Maintenance on the High Pressure Coolant Injection System
05000293/2008-005-02	NCV	Procedural Error Resulting in Unplanned RCIC Isolation

Closed

05000293/2008-003-00	LER	RCIC System Declared Inoperable During Surveillance Testing Due to Procedure Error
05000293/2007-003-04	URI	Application of TS 4.0.3 When it is Discovered that a Surveillance Has Never Been Performed

LIST OF DOCUMENTS REVIEWED

Section 1R01

Procedure 8.C.40, Revision 22, Seasonal Weather Surveillance
Procedure 2.2.35, Revision 42, Condensate Storage & Transfer System
Procedure 2.1.37, Revision 25, Coastal Storm Preparations and Actions
Procedure 2.1.42, Revision 7, Operation During Severe Weather

Section 1R04

Procedure 2.2.36, Revision 63, Instrument Air Systems
Instrument Air Drawings
Procedure 2.2.23, Revision 32, Automatic Depressurization System
EOP-01, Reactor Pressure Vessel Control
EOP-02, Reactor Pressure Vessel Control, Failure to Scram
UFSAR Section 10.4, Fuel Pool Cooling and Cleanup System
Procedure 2.2.85, Revision 73, Fuel Pool Cooling and Filtering System
Passport Database Equipment ID Printout on Spent Fuel Pool Pumps Suction Header Drain Valve
Procedure 1.17.1, Revision 9, Potential Seismic Interaction Hazards
Procedure 8.C.43, Revision 9, Monthly System Valve Lineup Surveillance
Drawing P&ID 212, Revision 91, Service Water System
Procedure 2.2.20, Revision 70, Core Spray System
UFSAR Section 6, Core Standby Cooling Systems
TS 3.5.A, Core Spray and Low Pressure Coolant Injection Systems
PNPS Training Manual Drawing, Core Spray System
P&ID M242, Core Spray System

Section 1R05

CR-PNP-2008-03122, Fire Protection Water Pipe Corroded Above Cable Trays
Fire Hazard Analysis, Fire Zone Data Sheet, Fire Area 1.9, Fire Zone 1.9, Reactor Building El. 23' to El. 51'/East Side
Fire Protection Engineering Evaluation (FPEE) 49, Revision 0, Barrier Between "A" Division Battery Room and Switchgear Room
FPEE 91, Revision 0, Battery Room Fire Doors
FPEE 92, Revision 0, III-T Penetration in Barrier between "A" Division Battery Room and Switchgear Room
FPEE 103, Revision 0, Battery Room/Switchgear Room Unfilled Block Walls
FPEE 45, Revision 2, Unfilled Block Walls, Recessed Electrical Boxes, Appendix A Barriers
FPEE 47, Revision 0, Turbine Building Floor 212.604 and Turbine Deck Storage Room Ceiling 196.603
FPEE 68, Revision 1, Cable Spreading Room Conduit/Removable Panel
FPEE 95, Revision 0, Type III-T Penetration Seal in Barrier 194.504A between "B" Switchgear Room and Stairway No. 8

FPEE 123, Revision 0, New Red Line Building – Exposure to Process Buildings
FPEE 126, Revision 0, Qualification of MTS-3 Installation on Enclosure No. 1
FPEE 127, Revision 0, Qualification of MTS-3 Installation on Enclosure No. 2
Procedure 8.B.17.2, Revision 9, Inspection of Fire Damper Assemblies
Fire Hazards Analysis Fire Zone Data Sheet Fire Area 1.9, Fire Zone 1.13, Fuel Pool Cooling Pumps/Heat Exchanger Area
Engineering Evaluation No. 86, Acceptability of Structural Steel Supporting Floor at E1.74 "B" of Reactor Building
Procedure 2.2.29, Revision 26, Smoke and Detection Systems
Fire Hazards Analysis Fire Zone Data Sheet, Fire Area 1.9, Fire Zone 3.5, Vital Motor Generator Set Room
Exemption Request No. 9, Fixed Fire Suppression with Alternate Shutdown Capability
Exemption Request No. 23, Walls with Ratings less than 3 Hours
Engineering Evaluation No. 19, Non-Fire Rated Materials in Seismic Joints
Engineering Evaluation No. 98, Cable Tray Penetration in Appendix "A" Barrier
Procedure 5.5.2, Revision 40, Special Fire Procedure

<u>Section 1R11</u>

Instructional Module, Revision 0, Core Damage/Mitigation Assessment
Severe Accident Guidelines
Severe Accident Guidelines LORT Overview – Fall 2008 Presentation Slides
Severe Accident Management Accident Phenomenology Presentation Slides

<u>Section 1R12</u>

CR-PNP-2008-02120, PASS Declared Inoperable
CR-PNP-2008-02121, H_2O_2 Declared Potentially Maintenance Rule (a)(1)
Regulatory Guide 1.160, Revision 2, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
NUMARC 93-01, Revision 2, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants
PASS System Health Report
Maintenance Rule (a)(1) Action Plan, Post Accident Sampling System
CR-PNP-02469, Standby Gas Treatment Root Valve 31-HO-8 does not fully shut
Procedure EN-DC-206, Revision 1, Maintenance Rule (a)(1) Process
Procedure EN-DC-207, Revision 1, Maintenance Rule Periodic Assessment
Functional Failure Determination Form on CR-PNP-2008-02469, Standby Gas Treatment
CR-PNP-2008-03338, HPCI Valve Undervoltage Trouble
HPCI System Health Report
Functional Failure Determination Form on CR-PNP-2008-03338, HPCI Undervoltage
K-117 Coil Degradation Apparent Cause Evaluation
K-117 (a)(1) Action Plan (CR-PNP-2008-03113)
CR-PNP-2008-03113, K-117 Air Compressor has exceeded Maintenance Rule Performance Criteria

<u>Section 1R13</u>

Equipment Out Of Service, Risk Assessment Tool
Procedure 1.5.22, Revision 11, Risk Assessment Process
Control Room Logs
Procedure 3.M.1-45, Revision 6, Outage Shutdown Risk Assessment
Risk Assessment Review Checklists
CR-PNP-2008-03356, RCIC Turbine Flow could not be adjusted to within procedure parameters
CR-PNP-2008-03792, No Risk Review performed of emergent HPCI failure

Section 1R15

CR-PNP-2008-3049, Air Void in HPCI Suction Line
Isometric Drawing MI00-256-1, High Pressure Coolant Injection
Generic Letter 2008-01, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems
ABS Consulting – Suction Void Calculations
GE BWR Owner's Group Technical Report ECCS Pumps Suction Void Fraction Study
CR-PNP-2008-03015, HPCI Cabling In-Service Aging Needs to be reviewed
Drawing E320, Revision E19, Turbine Building – Area 7, Conduit and Tray Layout
Drawing E318, Revision E13, Turbine Building – Area 6, Conduit and Tray Layout
CR-PNP-2008-03404, when clearing T/O 46A-017 and restoring F15 Circuit Switches and SDB, Breaker A802 would not close
Reasonable Expectation of Operability Form for CR-PNP-2008-3404
CR-PNP-2008-03446, After Change Out of A802 Control Switch, the A802 Breaker would not close from the Control Room
UFSAR Section 5.2.3.5.3, Instrument Piping Connected to the Reactor Primary System
CR-PNP-2007-04801, Restriction Orifices Missing from Recirculation Pump Instrument Lines
Operability Evaluation for CR-PNP-2007-04801
Drawing M251, Sheet 2, Revision 21, P&ID Recirculation Pump "B" Instrumentation
Calculation S&SA166, Revision 0, Instrument Line Break Blowdown
Calculation S&SA167, Revision 0, Reactor Building Response to Instrument Line Break
Calculation No. PNPS-1-ERHS-X111.Z-66, Revision 0, Radiological Impact Study of 1" RWCR Instrument Sensing Line Break inside RB Secondary Containment (with and without Restricting Orifice) and with or without SGTS.
CR-PNP-2008-03611, Thermography identifies a hot spot in ACB 104B
ODMI Implementation Action Plan for ACB 104B
CR-PNP-2008-01995, ACB 104B Disconnect Not Seating Properly
Procedure 3.M.3-60, Revision 6, Infrared Thermography
CR-PNP-2008-03669, Disconnect 102A has a misaligned blade

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EC Summary Report
ER 02115031, Change Orientation of PSV-8008
50.59 Screening Form, Change Orientation of PSV-8008
Engineering Change 7768, Temp Mod Required to Document Heat Detection System Disabled on Pre-Action Sprinkler System for the Turbine Bearings
Procedure EN-DC-136, Revision 3, Temporary Modifications
EN-LI-100, Revision 7, Attachment 9.1, Process Applicability Determination

UFSAR, Chapter 10.8, Revision 26, Fire Protection System
Procedure EN-DC-128, Revision 3, Fire Protection Impact Reviews
Report Number 89XM-1-ER-Q, Updated Fire Hazards Analysis
Procedure 2.2.26, Revision 40, Deluge Sprinklers and Spray Systems
CR-PNP-2008-03619, Control Room Drawing does not Reflect Temporary Modification Installation
Procedure 1.2.4, Revision 45, Operations Performance Assessment Program (OPAP)
CR-PNP-2008-03169, QA Identifies no audits of temporary modifications were performed by Operations during 2008

Section 1R19

CR-2008-03182, Unplanned RCIC Isolation and Turbine Trip
Procedure 8.Q.3-8.1, Revision 14, Limitorque Type HBC, SB/SMB-OO, and Type SMB-OOO Valve Operator Maintenance
WO 51665611, MOV Maintenance & Inspection
WO 5153695901, Open/Inspect RHR Pump Discharge Check Valve 1001-67A
10 CK-1001-67A Compliance Package
10 CK-1001-67A Quality Inspection Plan
Procedure 3.M.4-53, Revision 4, Check Valve Disassembly and Inspection
Vendor Manual V-0442, Velan Gate, Globe, Stop Check Valves
Drawing M132A8, Velan Forged Bolted Bonnet Swing Check Valve
WO 0016948901, Received Motor Control Center D9 Trouble Alarm. MO-2301-08 Position Lights Out
Procedure 3.M.3-51, Revision 26, Electrical Termination Procedure
Procedure 8.I.30, Revision 5, Operability Test for Valve Indicator Light Verification
Procedure 8.I.11.11, Revision 9, Reactor Coolant Pressure Boundary Isolation Valve Cold Shutdown Operability
Procedure 8.I.1.1, Revision 21, Inservice Pump and Valve Testing Program
WO 5167091201, MO-1001-34A Breaker Testing
Procedure 8.Q.3-3, Revision 54, 480V AC Motor Control Center Testing and Maintenance
CR-PNP-2008-03399, Calculation Error Identified MO-1001-34A Breaker Testing
WO 5168109201, M-1001-34A MOV Diagnostic Test
Procedure 3.M.3-24.16, Revision 11, Quicklook Operations Procedure
WO 5167091101, MO-1001-36A Breaker Testing
WO 5166675001, MO-1001-36A MOV Diagnostic Test
WO 5167011001, MO-1001-37A MOV Diagnostic Test
WO 5166999501, MO-1001-18A MOV Diagnostic Test
WO 5167107601, MO-1001-7A Breaker Testing
WO 5167091301, MO-1001-16A Breaker Testing
WO 5167091401, MO-1001-23A Breaker Testing
WO 5166675101, RHR Pump “A” Relay Testing
Procedure 3.M.3-1, Revision 123, A5/A6 Buses 4KV Protective Relay Calibration/Functional Test and Annunciator Verification
Drawing E5-200, Revision 7, 4160 Volt Switchgear Relay Settings
CR-PNP-2008-03262, As-Found Data Out of Specification for RHR “A” Relays
Procedure 1.3.34, Revision 15, Attachment 9, Surveillance Test Review for Procedure 3.M.3-1 dated 10/15/2008
Procedure 8.5.2.3, Revision 47, Low Pressure Coolant Injection and Containment Cooling Motor

Operated Valve Operability Test
Procedure 8.5.2.2.1, Revision 51, Low Pressure Coolant Injection System Loop “A” Operability – Pump Quarterly and Biennial (Comprehensive) Flow Rate Tests and Valve Tests
WO 00154061 05, Overhaul SSW P-208D in accordance with 3.M.4-14.2
Procedure 3.M.4-14.2, Revision 53, Salt Service Water Pumps; Routine Maintenance
Drawing M8-39, Revision 1, SSW Pump Bearing Retainer for Bronze-Backed Cutless Rubber Standard and Oversized O.D.
Drawing M8-38, Revision 7, SSW Pump Lineshafts
Drawing M8-4, Revision 27, Assembly Drawing Service Water Pump P208A, B, C, D, & E
WO 00154061 07, Overhaul SSW P-208D in accordance with 3.M.4-1
CR-PNP-2008-02675, P-208D Downstream Expansion Joint found out of tolerance
WO 00154036-01, Rebuild SSW P-208, Remove/Replace Expansion Joint, Replace 29-CK-3880D with Refurbished Valve, Inspect AV-38006
Procedure 3.M.3-4, Revision 53, Attachment 21, Insulation Test of 480V Related Loads and Cables
Procedure 3.M.3-51, Revision 26, Electrical Termination Procedure
Procedure 3.M.3-17.1, Revision 23, Raychem or Taping of 1000 Volt and Under Cables and/or wires
Procedure 3.M.1-14, Revision 22, General Maintenance Procedure for Heavy Load Handling Operations
Drawing E52A1, Revision E2, Outline and Dimension Salt Service Pump Motor
Drawing M212, Revision 91, P&ID Service Water System
WO 00154061 03, Post Maintenance Test P-208D
Procedure 8.5.3.2.1, Revision 21, Salt Service Water Pump Quarterly and Biennial (Comprehensive) Operability and Valve Operability Tests
Procedure 3.M.1-15, Revision 42, Vibration Monitoring for Preventive Maintenance and Balancing
Procedure ENN-NDE-10.02, Revision 3, VT-2 Examination
WO 00164079, Replace Three Relaying Current Transformers at F15
Schematic Meter and Relay Diagram 23KV Line and Shutdown Transformer
Product Bulletin for Type KOR-15C Current Transformer
Test Results by Omicron for Excitation Curve Data

Section 1R20

Procedure 1.3.37, Revision 27, Post Trip Review
Reactor Plant Event Notification Worksheet
CR-PNP-2008-03962, Reactor Scram Due to Switchyard Fault
CR-PNP-2008-03963, Following Reactor Scram, Y3 and Y4 buses were de-energized
Y-3/Y-4 Load List/Drawings
CR-PNP-2008-03965, Post Scram Reactor Water Sample
UFSAR Chapter 8.8, 120 VAC Power Systems
CR-PNP-2008-03967, Turbine Load Limit Remained at 100% following Turbine Trip and Reactor Scram
CR-PNP-2008-03968, Not all BTV’s closed on the Turbine Trip
CR-PNP-2008-03969, Relief/Safety Valve Leakage Alarms
Control Room Logs
Procedure 5.3.18, Revision 27, Loss of 120V AC Safeguard Buses Y3 and Y31
Procedure 5.3.19, Revision 30, Loss of 120V AC Safeguard Buses Y4 and Y41

Procedure 2.2.12, Revision 38, 120V AC Safeguard Power Supply: Y3-Y4, Y31-Y41, and Y13-Y14
Procedure 2.1.1, Revision 166, Startup from Shutdown
Procedure 2.1.4, Revision 26, Approach to Critical
Forced Outage Schedule
Forced Outage Action Item List

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Procedure 8.4.1, Revision 64, Standby Liquid Control Pump, Quarterly and Biennial Capacity and Flow Rate Test
TS 4.4, Standby Liquid Control System Surveillance Requirements
TS 3.13, In-Service Testing
CR-PNP-2008-3216, Inadvertent Release of SBLC Pump pushbutton during testing
Procedure 8.9.1, Revision 111, Emergency Diesel Generator and Associated Emergency BUS Surveillance
TS 4.9.A, Surveillance Requirements for Auxiliary Electrical Equipment
UFSAR Section 8.5, Standby AC Power Source

Section 2OS1

Procedure 6.1-009, Revision 15, Radiological Controls for Handling Highly Radioactive Objectives and Refuel Floor Activities

Section 2OS2

Procedure EN-RP-110, Revision 5, ALARA Program
Procedure 6.1-220, Revision 4, Radiological Controls for High Risk Evolutions
Pilgrim Station RFO 16 ALARA Report, April-May 2007

RWP Termination and Post Job ALARA Reviews:

07-0066, 07-0078, 07-0079, 07-0080, 07-0081, 07-0116, 07-0140

Section 4OA1

NRC Performance Indicator Data Sheet MSPI- Cooling Water System/RBCCW October 2007 – September 2008
NRC Performance Indicator Data Sheet MSPI- Cooling Water System/SSW October 2007 – September 2008
NRC Performance Indicator Data Sheet MSPI- Emergency AC Power/EDG October 2007 – September 2008
NEI-99-02, Revision 5, Regulatory Assessment Performance Indicator Guidelines

Section 4OA2

CR-PNP-2008-1102, SRV Leakage Alarm Received
CR-PNP-2008-5089, SRC-3C Tailpipe Temperature Indicates Rising Trend
CR-PNP-2007-4936, SRV Insulation

CR-PNP-2007-3432, SRV-3B Tailpipe Temperature
CR-PNP-2007-2920, As-Found SRV Test Results from Wyle Labs
CR-PNP-2007-0143, Target Rock As-Found Test Results
CR-PNP-2004-1368, As-Found Setpoint Testing at Wiley Labs
EE 01-022, Engineering Evaluation for Target Rock Corporation Two-Stage Safety Relief Valve 203-3C, Rev. 1, dated 04/12/01
Procedure 2.2.23, Revision 32, Automatic Depressurization System
EN-LI-102, Revision 12, Corrective Action Process
Entergy Quality Assurance Program Manual, dated 04/15/08
Automatic Depressurization System Reference Text, Rev. 4
NEDE-33110, BWROG SRV Leakage Reduction, Rev. 0, Class 1, July 2003
General Electric (GE) Service Information Letter (SIL) 196, Summary of Recommendations for Target Rock Main Steam Safety Relief Valves, dated 09/30/76
GE SIL 196, Supplement 11, Recommendations Applicable to the Target Rock Main Steam Safety Relief Valve Model #7567F (Two-Stage SRV Design), April 1982
GE SIL 196, Supplement 13, Target Rock SRV Base to Body Flange Joint Leakage, November 1983
GE SIL 196, Supplement 14, Target Rock Two-Stage SRV Setpoint Drift, dated 04/23/84
GE SIL 196, Supplement 16, Target Rock SRV Insulation Maintenance, dated 09/03/92
License Amendment No. 56, Incorporation of LCOs and SRs for SRV Discharge Piping, dated 03/20/82
License Amendment No. 73, SRV Setpoints, dated 03/26/84
License Amendment No. 222, Deletion of Requirement for NRC Approval of Engineering Evaluation of SRV Operability when SRV Discharge Pipe Temperature Exceeds 212ºF, dated 09/27/06
NRC Safety Evaluation Related to Amendment No. 222, dated 08/04/06
NRC Regulatory Issue Summary 200-12, Resolution of Generic Safety Issue B-55, "Improved Reliability of Target Rock Safety Relief Valves, dated 08/07/00
NRC Internal MEMO: Closeout of Generic Safety Issue B-55, Improved Reliability of Target Rock Safety Relief Valves (ML993620214), dated 12/17/99
NRC Information Notice 80-25, Operating Problems with Target Rock Safety-Relief Valves at BWRs, dated 12/19/80
NUREG-1022, Event Reporting Guidelines, Rev. 2
NRC IN 83-82, Failure of Safety Relief valves to Open at BWR – Final Report, dated 12/20/83
NRC IN 86-12, Target Rock Two-Stage SRV Setpoint Drift, dated 02/25/86
NRC IN 88-30, Target Rock Two-Stage SRV Setpoint Drift Update, dated 05/25/88
Technical Specifications Section 3.6, Primary System Boundary
UFSAR 4.4, Nuclear System Pressure Relief
UFSAR 6.1 – 6.5, Core Standby Cooling Systems
Vendor Manual No. V0353, Target Rock Safety Relief Valve Model 7567F, dated 07/29/08
Procedure 1.3.34.4, Revision 17, Compensatory Measures
Pilgrim Operator Workarounds Aggregate Report
Compensatory Measures and Disabled Annunciater Logs
Procedure 5.3.35.1, Revision 4, Transient Response Hardcards for Operating Crews
Procedure 2.2.21, Revision 73, HPCI System
Procedure 2.2.22, Revision 69, RCIC System
CR-PNP-2007-04865, Review of 2007 events
Operations Night Orders TEAR dated 11/24/2008

Licensing Training for Engineering Staff
Procedure EN-LI-118, Revision 7, Effectiveness Review Criteria
Pilgrim Station Quarterly Trend Report for Third Quarter 2008
Mentorship Pilot Program
CR-PNP-2007-03925, Mispositioning Adverse Trend
CR-PNP-2007-03036, Mispositioning of Exhaust Fan Control Switches

Section 4OA3

Event Notification Sheet No. 44545
Procedure 8.M.2-2.6.3, RCIC Steam Line High Temperature
Control Room Logs
Event Timeline
Plymouth Fire Department Fire Investigation Summary
Incident Report of Emergency Declared at Pilgrim Nuclear Power Station
Radiological Survey Forms
Procedure EP-IP-100, Revision 29, Emergency Classification and Notification
Root Cause Analysis Report for CR-PNP-2008-03433
CR-PNP-2008-03433, Fire in HP Calibration Room
Fire in HP Calibration Room and Battery Use/Storage Precautions Tailgate Message
Technical Specifications
Emergency Action Level Chart
10 CFR 50.72 Event Notification Worksheet dated 10/22/2008
CR-PNP-2008-03356, RCIC flow and pressure unable to be adjusted to within test parameters
CR-PNP-2008-03288, RCIC inoperable due to power supply capacitors age significantly exceeds recommended limits
CR-PNP-2008-03962, Reactor Scram Due to Switchyard Fault
CR-PNP-2008-03963, Y3 and Y4 Busses were Deenergized
CR-PNP-2008-04003, Condensate Pump Recirculation Valve Failure Inhibits Plant Startup
CR-PNP-2008-04086, Review of Plant Trip Identifies Negative Performance of Turbine Controls
PNPS December Forced Outage Startup Chart
Procedure 5.3.18, Revision 27, Loss of 120V AC Safeguard Buses Y3 and Y31
Procedure 5.3.19, Revision 30, Loss of 120V AC Safeguard Buses Y4 and Y41
Procedure 2.2.12, Revision 38, 120V AC Safeguard Power Supply: Y3 – Y4, Y31 – Y41, and Y13 – Y14
Procedure 1.3.37, Revision 27, Post-Trip Reviews

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Reasonably Achievable
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator

EOP	Emergency Operating Procedure
FPEE	Fire Protection Engineering Evaluation
HP	Health Physics
HPCI	High Pressure Coolant Injection
IR	Inspection Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
O&M	Operations and Maintenance
PARS	Publicly Available Records
PASS	Post Accident Sample System
PI	Performance Indicator
PMT	Post Maintenance Test
PNPS	Pilgrim Nuclear Power Station
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RHR	Residual Heat Removal
RWP	Radiation Work Permit
SAG	Severe Accident Guideline
SBLC	Standby Liquid Control
SCBA	Self Contained Breathing Apparatus
SDP	Significance Determination Process
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SSW	Salt Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report