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

July 28, 2011

Mr. Robert Smith
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 05000293/2011003

Dear Mr. Smith:

On June 30, 2011, 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 July 14, 2011, with you and other 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.

This report documents five NRC-identified findings of very low safety significance (Green). Four of these findings were determined to be violations of NRC requirements. However, because of the very low safety significance and because they have been entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section 2.3.2.a 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 cross-cutting aspect assigned to 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.

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

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Sincerely,

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

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

Enclosure: Inspection Report 05000293/2011003
w/Attachment: Supplemental Information

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

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-293
License No:	DPR-35
Report No:	05000293/2011003
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Pilgrim Nuclear Power Station (PNPS)
Location:	600 Rocky Hill Road Plymouth, MA 02360
Inspection Period:	April 1, 2011 through June 30, 2011
Inspectors:	M. Schneider, Senior Resident Inspector, Division of Reactor Projects (DRP) B. Smith, Resident Inspector (DRP) K. Dunham, Acting Resident Inspector (DRP) R. Rolph, Health Physicist, Division of Reactor Safety (DRS) T. Burns, Senior Reactor Inspector (DRS)
Approved By:	Donald E. Jackson, Chief Projects Branch 5 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000293/2011003; 04/01/2011-06/30/2011; Pilgrim Nuclear Power Station; Fire Protection, Flood Protection Measures, Maintenance Risk Assessments and Emergent Work Control, Operability Evaluations.

The report covered a three-month period of inspection by the resident and regional-based inspectors. Four non-cited violations (NCVs) and one finding of very low safety significance (Green) 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." Cross-cutting aspects associated with findings are determined using IMC 0310, "Components Within the Cross-Cutting Areas." 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.

NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a Green finding (FIN) for the improper maintenance of underground non-safety related medium voltage electric cables. The inspectors identified that Entergy allowed non-safety related medium voltage cables to remain submerged in water for extended periods of time. Entergy entered this issue into their corrective action program, specified corrective actions to increase the dewatering frequency of the affected manhole, and then installed an automatic dewatering pump.

  The inspectors determined that the finding was more than minor because it was associated with the Design Control attribute of the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, continued submergence of the non-safety related power cables (from the start-up transformer to electrical buses A2 and A4) could lead to cable failure and cause an event that would affect plant stability. The inspectors performed a Phase 1 Significance Determination Process screening of the finding in accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance because the condition did not contribute to both the likelihood of a reactor trip and the unavailability of mitigating systems equipment.

  The inspectors determined this finding had a cross-cutting aspect in the Problem Identification and Resolution cross-cutting area, Corrective Action Program component, because Entergy personnel did not implement corrective actions in a timely manner to ensure that underground cables were not submerged, commensurate with the safety significance and complexity of the degraded condition [P.1(d)]. (Section 1R06)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green NCV of License Condition 3.F of the Pilgrim Facility Operating License (DPR-35) for the failure to evaluate transient combustible fire loading in the Standby Liquid Control (SLC) room. Specifically, Entergy did not evaluate the acceptability of transient combustibles that had been moved into the SLC room which were in excess of the allowed combustible loading discussed in the Fire Hazards Analysis. Entergy immediately walked down the area, established compensatory measures, and completed a transient combustibles evaluation. Entergy has since removed the transient combustibles from the area.

  The inspectors determined that the failure to evaluate the transient combustibles was more than minor based on a similar example described in Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," Section 4k. Specifically, the fire loading exceeded the Fire Hazard Analysis assumption and was not evaluated for acceptability. The finding is also associated with the Protection Against External Events attribute of the Mitigating Systems cornerstone and could have adversely affected the cornerstones objective to ensure the availability of systems that respond to events to prevent undesirable consequences (i.e., core damage). Specifically, a fire in the SLC room could affect the availability of the SLC system to respond to an event. IMC 0609, "Significance Determination Process," Appendix F, "Fire Protection Significance Determination Process," was used to evaluate the significance of the finding. The safety significance of the finding was determined to be very low because the degradation factor was low; that is, the transient combustible evaluation process subsequently identified nearly the same level of fire protection effectiveness and reliability for the SLC room as it would have if the degradation had not been present.

  This finding had a cross-cutting aspect in the Human Performance cross-cutting area, Work Control component; in that, Entergy did not coordinate work activities to ensure the interdepartmental coordination necessary to assure plant and human performance. Specifically, the refueling organization did not notify fire protection engineering to ensure an evaluation of the transient combustible loading was completed for the SLC room [H.3(b)]. (Section 1R05)

- Green. The inspectors identified a Green NCV of 10 CFR 50.65 paragraph (a)(4) for Entergy's failure to conduct an adequate risk assessment for planned Analog Trip System (ATS) testing. Specifically, the inspectors identified that Entergy had not analyzed the impact to the risk of the plant with a reactor low pressure master trip unit removed from service during the ATS test. The removal of this instrument resulted in an Orange risk condition. Entergy has implemented corrective actions to revise the risk assessment procedure to provide specific guidance on assessing surveillance procedures which affect multiple components; established guidance to complete risk assessment reviews six weeks prior to the scheduled performance of planned work and test activities; and provided guidance and training on the above to personnel involved in the risk assessment process.

  The inspectors determined that this issue was more than minor because the actual overall plant risk was in a higher licensee-established risk category (Orange) than what Entergy had previously determined (Yellow). Entergy's risk assessment had incorrect assumptions that changed the outcome of the assessment. The inspectors performed a screening 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 the reactor low pressure instrument was removed from service was less than 1E-6 (approximately 1E-8).

This finding had a cross-cutting aspect in the Human Performance cross-cutting area, Decision Making component, because Entergy did not use a systematic process to make a risk-significant decision [H.1(a)]. (Section 1R13)

- Green. The inspectors identified a Green NCV of Technical Specification (TS) 3.3.B.1 "Control Rod Operability," for Entergy's failure to enter and perform the actions prescribed in Technical Specifications after losing control rod position indication. Entergy has since restored control rod position indication by repairing a failed power supply. Condition report CR-PNP-2011-0272 was written to address the power supply failure and condition report CR-PNP-2011-0511 was subsequently written to address Entergy's administration of TSs.

  The inspectors determined that the issue 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 reliability of systems that respond to events to prevent undesirable consequences (i.e., core damage). Specifically, the locations of the control rods were indeterminate which could substantially impact operator's abilities to implement Emergency Operating Procedures. IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1-Initial Screening and Characterization of Findings," was used to evaluate the significance of the finding. Attachment 0609.04, Table 4a, was used to evaluate the impact of the finding on loss of operability or functionality. The inspectors determined that the function of the control rods to add negative reactivity to the core during an event was not affected (SCRAM time and control rod worth were not affected). In addition, alternate means were available to operators to determine control rod position and once the power supply was restored, the control rods were determined to have remained in their original positions. Also, since the finding is not potentially risk significant due to a seismic, flooding or severe weather initiating event, the finding was determined to be of very low safety significance (Green).

  The inspectors determined that this issue had a cross-cutting aspect in the Decision Making component of the Human Performance cross-cutting area. Specifically, Entergy did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disprove the action. In this case, Entergy did not take the conservative approach to enter Technical Specifications when faced with a degraded condition affecting control rod operability [H.1(b)]. (Section 1R15)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green NCV of Technical Specification (TS) 3.7.B.2.f, "Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (CRHEAFS)," for Entergy's failure to enter and perform the actions prescribed in TS after the Control Room Envelope (CRE) was breached during work on a vital area door into the CRE. Entergy has since repaired the vital area door and restored the CRE.

This finding was more than minor because it was associated with the Human Performance attribute of the Barrier Integrity cornerstone (maintain the radiological barrier function of the control room) and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the inoperable CRE could affect the operator's ability to occupy the control room under adverse radiological, chemical, or smoke conditions while responding to an event. IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1- Initial Screening and Characterization of Findings," was used to evaluate the impact of the finding on loss of operability or functionality of the CRE and CHREAFS, and it was determined that further evaluation was required since the finding had the potential to impact the control room envelope due to the effects of smoke and toxic gas. As a result of this screening, a Phase 3 evaluation was conducted by a Senior Reactor Analyst (SRA). The SRA conducted a qualitative evaluation and determined the risk impact on control room habitability, due to this finding, from smoke and toxic gas to be low (Green). Specifically, the Pilgrim Station Individual Plant Examination for External Events (IPEEE), sections 5.3.3 and 5.3.4, identified that the overall risk from on-site and off-site chemical release was low.

The inspectors determined that this issue had a cross-cutting aspect in the Work Control component of the Human Performance cross-cutting area. Specifically, Entergy did not plan and coordinate work activities affecting the CRE such that interdepartmental coordination assured plant and human performance. In this case, Operations was not made aware that Maintenance would be working on the control room vital door [H.3(b)]. (Section 1R15)

REPORT DETAILS

Summary of Plant Status

Pilgrim Nuclear Power Station (PNPS) began the inspection period operating at 90 percent reactor power in an end-of-cycle power reduction, and was shutdown on April 17, 2011, to begin refueling outage (RFO) 18. Following completion of RFO 18 activities, a reactor start up was performed on May 10, 2011; however, the plant scrammed on high intermediate range monitor indication. Operators investigated the event, commenced another start-up on May 11, 2011, and achieved 15 percent reactor power. The plant was returned to a shutdown condition on May 14, 2011, due to the inability to establish an acceptable torus to drywell differential pressure. On May 18, 2011, a reactor start up was performed and the plant reached 100 percent reactor power on May 21, 2011. On May 21, 2011, operators reduced power to 87 percent reactor power to perform a control rod pattern adjustment and returned to 100 percent reactor power later that same day. On May 22, 2011, operators reduced power to 76 percent reactor power to perform a control rod pattern adjustment. Operators restored the plant to 100 percent reactor power on May 23, 2011. On June 15, 2011, Operators reduced power to 77 percent reactor power to perform a control rod pattern adjustment, returned to 100 percent reactor power later that same day, and operated at or near 100 percent reactor 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 Seasonal Susceptibility

a. Inspection Scope (1 sample)

The inspectors performed a review of severe weather preparations during the week of May 16, 2011, to evaluate the site's readiness for the onset of hurricane season, including the readiness of three risk-significant systems: the intake structure; the Emergency Diesel Generators (EDGs); and the Station Blackout (SBO) Diesel Generator. The inspection examined selected equipment and supporting structures to determine if they were configured in accordance with Entergy procedures and if adequate controls were in place to ensure functionality of the systems. The inspectors performed partial walkdowns of the intake structure, the EDG enclosures, and the SBO enclosure to determine the adequacy of equipment protection from the effects of hurricanes. The documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings were identified.

.2 Alternating Current (AC) Power System Readiness

a. Inspection Scope (1 sample)

The inspectors performed a review of Entergy's offsite and alternate AC power system readiness for susceptibilities during adverse weather. The inspectors reviewed Entergy's plant features and procedures for operation and continued availability of offsite and alternate AC power systems to determine if they were appropriate. The inspection focused on procedures affecting these areas and communication protocols between the transmission system operator (TSO) and Entergy to verify that appropriate information would be exchanged when issues arise that could impact the offsite power system. The inspectors also reviewed Entergy's procedures to ensure that they addressed actions to be taken when notified by the TSO to transfer safety-related loads to the onsite power supply; compensatory actions to be performed if it were not possible to predict grid conditions; reassessment of plant risk based on maintenance activities which could affect grid reliability; and required communications between Entergy and the TSO. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns (71111.04Q)

a. Inspection Scope (3 samples)

The inspectors performed three partial system walkdowns during this inspection period. 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 procedures, and to identify any discrepancies that may have had an effect on operability. The walkdowns included selected control switch position verifications, valve position checks, and verification of electrical power to critical components. In addition, the inspectors evaluated other elements, such as material condition, housekeeping, and component labeling. The documents reviewed are in the Attachment. The following systems were reviewed based on their risk significance for the given plant configuration:

- 'B' Emergency Diesel Generator (EDG) following outage maintenance;
- Standby Liquid Control during startup after a prolonged outage; and
- Station Blackout Diesel Generator with 'A' EDG in a degraded condition.

b. Findings

No findings were identified.

.2 Complete System Walkdowns (71111.04S)

a. Inspection Scope (1 sample)

The inspectors completed a detailed review of the Fuel Pool Cooling (FPC) 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 and breakers were aligned in accordance with operating procedures, and to assess the material condition of valves and other supporting equipment. The inspectors discussed system health with the system engineer, reviewed the system's Maintenance Rule status, and performed a review of outstanding maintenance work orders to determine whether the deficiencies significantly affected the FPC system function. The inspectors also reviewed condition reports from the past year to determine whether FPC equipment problems were being identified and appropriately resolved. The documents reviewed are listed in the Attachment.

b. Findings

No findings 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 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 1.30, Fire Zone 1.30, Drywell;
- Fire Area 1.9, Fire Zone 2.2, 'A' Switchgear and Load Center Room;
- Fire Area 1.9, Fire Zone 1.15, Standby Liquid Control System Room;
- Fire Area 1.10, Fire Zone 1.23, Standby Gas Treatment System Room; and
- Fire Area 1.10, Fire Zone 4.1, 'B' Emergency Diesel Generator Room.

b. Findings

Introduction: The inspectors identified a Green NCV of License Condition 3.F of the Facility Operating License (DPR-35) for the failure to evaluate transient combustible fire loading for the Standby Liquid Control (SLC) room. Specifically, Entergy did not evaluate the acceptability of transient combustibles that had been moved into the SLC room which were in excess of the allowed combustible loading discussed in the Fire Hazards Analysis.

Description: During a fire protection walk down of the SLC room, the inspectors noted that monitoring equipment being used to support refueling outage activities was present without a fire watch and without a Transient Combustible Evaluation (TCE).

The inspectors contacted the fire protection engineer since the material appeared to be in excess of the 100 pounds of ordinary combustibles allowed in Level 3 plant areas as described in Entergy procedure EN-DC-161, "Control of Combustibles." The fire protection engineer walked down the room and determined that the combustible loading present required a TCE and an hourly fire watch. The transient combustibles listed on the TCE had a total heat value of 4,860,000 BTU, while the allowed transient combustible total heat value listed in Appendix A of the Fire Hazards Analysis is 4,000,000 BTU. The transient combustibles have since been removed from the room and the transient combustible loading was brought below the allowed loading discussed in the Fire Hazards Analysis.

Analysis: The performance deficiency was the failure to implement all provisions of the approved fire protection program. Specifically, Entergy did not evaluate the transient combustible loading for the SLC room per procedure EN-DC-161 "Control of Combustibles." This performance deficiency was reasonably within Entergy's ability to foresee and correct. Traditional enforcement does not apply because there were no actual safety consequences, no impacts on the NRC's ability to perform its regulatory function, and no willful aspects associated with the issue. The inspectors determined that the failure to properly evaluate transient combustible fire loading was more than minor based on a similar example in Inspection Manual Chapter 0612, "Power Reactor Inspection Reports", Appendix E, "Examples of Minor Issues," Section 4k. Specifically, the fire loading exceeded the Fire Hazards Analysis assumption and was not evaluated for acceptability. The finding is associated with the Protection Against External Events attribute of the Mitigating Systems cornerstone and could have adversely affected the cornerstone's objective to ensure the availability of systems that respond to events to prevent undesirable consequences (i.e., core damage). Specifically, a fire in the SLC room could affect the availability of the SLC system to respond to an event. The inspectors evaluated the significance of the finding in accordance with Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process." The safety significance of the finding was determined to be very low because the degradation factor was low; that is, the transient combustible evaluation process subsequently identified nearly the same level of fire protection effectiveness and reliability for the SLC room as it would have if the degradation had not been present.

This finding had a cross-cutting aspect in the Human Performance cross-cutting area, Work Control component; in that, Entergy did not coordinate work activities to ensure the interdepartmental coordination necessary to assure plant and human performance. Specifically, the refueling organization did not notify fire protection engineering to ensure an evaluation of the transient combustible loading was completed for the SLC room [H.3(b)].

Enforcement: License condition 3.F of the Pilgrim Facility Operating License (DPR-35), specifies that Entergy shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report. Additionally, the Fire Hazards Analysis accounts for a transient combustible loading of 4,000,000 BTU in the SLC room. Contrary to the above, Entergy failed to appropriately implement the approved fire protection program when they allowed a combustible loading of 4,860,000 BTU in the SLC room without a fire protection evaluation for acceptability. Entergy has entered this issue into the corrective action program (CR-PNP-2011-02149), completed an evaluation, established compensatory measures, and has since removed the transient combustibles from the area. Because this finding is of very low safety significance and has been entered into the corrective action program, it is being treated as an NCV, consistent with the NRC's Enforcement Policy. **(NCV 05000293/2011003-01, Transient Combustible Loading in SLC Room in Excess of the Fire Hazards Analysis Limit)**

1R06 Flood Protection Measures (71111.06)

Internal Flooding Inspection

a. Inspection Scope (1 sample)

The inspectors reviewed a sample of flood protection measures affecting cables located in underground manholes. The inspectors selected cable pits '2A', '26B', and 'K' that contain underground safety and non-safety related power cables. The inspectors monitored Entergy's maintenance inspection and dewatering activities associated with each manhole to evaluate the as-found condition and corrective actions. The inspectors assessed the condition of power cables, splices, and supports. The inspectors also reviewed Entergy's Cable Reliability Program. The documents reviewed are listed in the Attachment.

b. Findings

Introduction: The inspectors identified a Green finding (FIN) for improper maintenance of underground non-safety related medium voltage electric cables. The inspectors observed partially submerged medium voltage cables during an inspection of three cable vaults.

Description: The electric power distribution system provides power to safety and non-safety related distribution buses in the plant. Off-site power is provided to the system by two independent circuits through non-safety related, medium voltage (typically those rated from 2 kilovolts to 35 kilovolts), Kerite cables that are routed through underground vaults and ducts. These cables are not rated for continuous submergence in water.

On June 9, 2011, the inspectors observed water in a manhole and vault containing the start-up transformer cables. The inspectors noted that no automatic dewatering or drainage systems existed in the manhole. Entergy procedure EN-DC-346, Revision 1, "Cable Reliability Program," discusses manhole inspections and dewatering, and requires, in part, "If manual inspections and pumping are used to maintain a cable system dry, the intervals must be sufficient to keep the cables dry. Adjust intervals as necessary, based on inspection results." Discussions with Entergy personnel involved with these inspections indicated that cables in Manhole '2A' had been found dry during the prior two inspections; however, the inspectors noted they were partially submerged during this inspection. The cables that were submerged included 4160V cables that are installed from the startup transformer and connected to the 'A2' and 'A4' non-safety related busses. The inspectors noted that Entergy had previously identified submerged cables in August and September of 2009 and in April of 2010; however, corrective actions have not been sufficient to preclude these cables from being submerged. The inspectors also noted that a finding had been issued in 2010 (FIN 05000293/2010003-

01, Submerged Medium Voltage Cables) which documented the failure to keep cables located in manhole '2A' dry.

Entergy generated Condition Report (CR) CR-PNP-2011-2911, and specified actions to increase the frequency of the dewatering activities for Manhole '2A' and has since installed an automatic dewatering system in Manhole '2A.'

Analysis: The inspectors determined that allowing medium voltage cables to remain submerged for extended periods of time was a performance deficiency. The cause of the issue was within Entergy's ability to foresee and correct, and should have been prevented. 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.

The finding was more than minor because it was associated with the Design Control attribute of the Initiating Events cornerstone, and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, continued submergence of the non-safety related power cables (from the start-up transformer to electrical buses A2 and A4) could lead to cable failure and cause an event that would affect plant stability. The inspectors performed a Phase 1 Significance Determination Process screening of the finding in accordance with NRC Inspection Manual Chapter 0609, Attachment 4, "Phase 1 -Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because the condition did not contribute to both the likelihood of a reactor trip and the unavailability of mitigating systems equipment.

The inspectors determined that this finding had a cross-cutting aspect in the Problem Identification and Resolution cross-cutting area, Corrective Action Program component, because Entergy personnel did not implement appropriate corrective actions in a timely manner to ensure that underground cables are not submerged, commensurate with their safety significance and the complexity of the degraded condition [P.1(d)].

Enforcement: No violation of regulatory requirements occurred. This finding does not involve enforcement action because no violation of a regulatory requirement was identified. Entergy has taken corrective actions to install an automatic dewatering device in the affected manhole. Because this finding does not involve a violation and has very low safety significance, it is identified as a finding (FIN). **FIN 05000293/2011003-02, Submerged Medium Voltage Cables.**

1R08 In-service Inspection (71111.08)

a. Inspection Scope (1 sample)

The purpose of this inspection was to assess the effectiveness of Entergy's In-service inspection (ISI) program for monitoring degradation of reactor pressure vessel internals, reactor coolant system boundary, risk significant piping system boundaries, and the containment boundary. The inspector assessed the inservice inspection activities using requirements and acceptance criteria for component examination specified in the

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, and applicable NRC Regulatory Requirements.

The inspector selected a sample of nondestructive examination (NDE) activities and performed a review to assess those activities for compliance with the requirements of ASME Section XI and applicable regulatory requirements. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation could result in a significant increase in risk of core damage in the event of loss of structural or pressure retaining capability.

The inspector verified by documentation review that test procedures and examiner qualifications were current and in accordance with the ASME Code requirements. Also, the inspector reviewed examiner qualifications for use of the performance demonstration initiative (PDI) manual and automatic ultrasonic test (UT) procedures to examine welds. The inspector selected a sample of Condition Reports (CR), operability determinations and corrective actions for review of Entergy's effectiveness in the identification and resolution of relevant indications discovered during ISI activities. The inspector's review of selected samples of non-destructive testing (NDE) included the following:

1. Ultrasonic testing, manual PDI-UT, phased array of dissimilar metal welds of recirculation, core spray and RPV nozzle to safe end reactor coolant pressure boundary welds.

2. Magnetic particle test of weld of reactor pressure vessel (RPV) head to vessel flange, RPV-HF, data sheet MT-001, and data report R-008.

3. VT-1 and VT-3 examination of in vessel components consisting of steam separator lifting lugs, guide rod brackets, tie rod upper support, jet pump main wedges, steam dryer leveling screw tack welds, and various welds of in vessel core spray piping and structural components.

4. Liquid penetrant test of weld repair of Reactor Building Closed Cooling Water (RBCCW) heat exchanger (E209A) channel wall. Liquid penetrant test performed in accordance with ASME Section XI.

The inspector reviewed selected in-vessel components and structural members to evaluate examiner skill, test equipment performance, examination technique, and inspection environment. The inspector also reviewed indications of intergranular stress corrosion cracking (IGSCC) noted in the steam separator lifting lugs. Because indications were noted in the base material and in the heat affected zone of fabrication welds (Indication Notification Report, PIR18-IVVI-11-06R1), the inspector reviewed the identification and characterization of the indications and the analytical analysis performed by the component vendor. The inspector reviewed the analysis which supported an "accept as-is" disposition for this outage. Also, indications reported during this examination in the steam dryer leveling screws and the tack weld in the lower partition plate were evaluated by comparison with previous examination results noted in 2009. The comparison revealed no noticeable change in characterization of the indications.

The inspector selected three ASME Section XI repair/replacement plans for review where welding was performed. The review was performed to confirm that appropriate weld procedures and welders were assigned this work. The inspector reviewed base materials and weld filler metal to determine if they were in accordance with Code requirements. The inspector determined the qualifications were in compliance with the requirements of ASME Section XI and IX. Also, the inspector reviewed documentation that the weld examinations were performed in accordance with the ASME code requirements. The three ASME Section XI repair/replacement work order (WO)/work request (WR) reviewed were:

1. WR 00274898: A welded connection on service water support H29-1-2 was found failed during routine system examination. CR-PNP-2011-1986 was initiated to document the as found condition. The weld was installed to drawing requirements (M8673) and nondestructively tested. No recordable indications were identified.

2. WO 52240225-19: The RBCCW heat exchanger (E209A) channel walls on one "end bell" required repair due to leakage between channels. The repair consisted of depositing sufficient copper-nickel weld metal to rebuild the channel locations. Weld repairs were liquid penetrant tested using procedure CEP-NDE-0640, Rev. 6. No recordable indications were identified.

3. WO 00249616-01: Modification to install two vent valves in high pressure coolant injection (HPCI) suction piping. Vent valves were installed in HPCI by welding in accordance with the requirements of ASME Section XI.

The inspector performed a review of the visual inspection results of portions of the primary containment and additional structural members attached to the liner to assess the condition of the protective coating. The inspector performed this visual review to determine the extent of any peeling, blistering, coating loss or other damage as a result of corrosion, foreign material impact or lack of maintenance. In addition, the inspector performed a visual evaluation of the interior surfaces of the torus including various structural members that were accessible from the catwalk to assess the condition of the protective coating. The inspector noted that previous visual inspection by Entergy examiners had identified various locations to monitor for comparison at the next outage. This inspection was in accordance with the requirements of ASME Section XI, IWE.

b. <u>Findings</u>

No findings were identified.

1R11 <u>Licensed Operator Requalification Program (71111.11)</u>

<u>Resident Inspector Quarterly Review</u> (71111.11Q)

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

The inspectors observed licensed operator performance during a licensed operator requalification training simulator exercise on June 9, 2011. The inspectors observed crew response to a loss of Turbine Building Closed Cooling Water and High Pressure

Coolant Injection steam leak scenario. 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 scenarios 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. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope (1 sample)

The inspectors reviewed the Intermediate Range Monitor System Performance Review as a sample for this inspection activity. The sample was reviewed 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 are listed in the Attachment.

b. Findings

No findings were identified.

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

a. Inspection Scope (7 samples)

The inspectors evaluated seven maintenance risk assessments for planned testing and 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 documents reviewed during this inspection are listed in the Attachment. The inspectors reviewed the conduct and adequacy of maintenance risk assessments for the following maintenance and testing activities:

- Orange Risk during maintenance on Vital Bus Breaker A501;
- Yellow Risk during Reactor Core Isolation Cooling maintenance, Standby Liquid Control testing, and Analog Trip System testing;
- Orange Risk during the outage due to limited decay heat removal availability;
- Yellow Risk during 'B' Emergency Diesel Generator maintenance and testing;
- Yellow Risk during maintenance on the K-117 Air Compressor and testing of the Station Blackout (SBO) Diesel Generator;
- Yellow Risk during maintenance and testing of the SBO, K-117 Air Compressor, and 'A' Reactor Protection System channel; and
- Green Risk during maintenance on the 'A' Control Rod Drive system.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR 50.65 paragraph (a)(4) for Entergy's failure to conduct an adequate risk assessment for planned Analog Trip System (ATS) testing. Specifically, the inspectors identified that Entergy had not analyzed the impact to the risk of the plant with a reactor low pressure master trip unit removed from service during the ATS test. The removal of this instrument resulted in an Orange risk condition.

Description. On April 4, 2011, Entergy intended to conduct planned maintenance on the Reactor Core Isolation Cooling (RCIC) system, and testing on both the Standby Liquid Control (SLC) system and the Analog Trip System (ATS). Entergy evaluated the plant risk and determined that the plant would be in a Yellow risk condition with this equipment out of service.

The inspectors independently performed a review of plant risk using the Equipment Out-Of-Service (EOOS) risk model. The inspectors determined from a review of the ATS test that a reactor low pressure master trip unit would be removed from service during the test. The inspectors then removed this component from service, along with the RCIC and SLC systems, in the EOOS risk model which calculated an Orange risk condition. The inspectors notified the control room of the difference between Entergy's risk assessment and the inspector's results. Operations decided to continue with the RCIC maintenance and the ATS test separately from the SLC test as a risk management action but did not update the risk profile to Orange. The inspectors then noted that the ATS component alone would cause the plant to be in an Orange risk condition. The inspectors again notified the control room. Operations decided to continue with the ATS testing and they did not update the advertised risk condition to Orange nor did they implement any further risk management actions.

Analysis. The performance deficiency associated with this finding is that Entergy incorrectly assessed the risk impact of RCIC maintenance, SLC testing, and ATS testing. In addition, Entergy did not update the risk nor take additional risk management actions when this was brought to their attention by the inspectors. The inspectors determined that this finding was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and 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 unavailability of a reactor low pressure master trip unit during the ATS test resulted in the plant being in an increased risk condition. The inspectors determined that this issue was more than minor because the actual overall plant risk was in a higher licensee-established risk category (Orange) than what Entergy had previously determined (Yellow). Entergy's risk assessment had incorrect assumptions that changed the outcome of the assessment. The inspectors performed a screening 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 the reactor low pressure instrument was removed from service was less than 1E-6 (approximately 1E-8). This finding had a cross-cutting aspect in the Human Performance cross-cutting area, Decision Making component, because Entergy did not use a systematic process to make a risk-significant decision. [H.1(a)]

Enforcement. 10 CFR 50.65 paragraph (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, on April 4, 2011, Entergy did not adequately assess the risk that resulted from the removal of a reactor low pressure master trip unit from service. In addition, Entergy did not upgrade the advertised plant risk and did not specify risk management actions for the increased risk condition. Corrective actions taken for this violation included revising the risk assessment procedure to provide specific guidance on assessing surveillances affecting multiple components; establishing guidance to complete risk assessments prior to six weeks before the work and testing is scheduled to be performed; revising the risk assessment procedure to provide guidance to work week managers and operations work liaisons to review the impact of work items on components with risk significance; and to conduct training on the above. Because this violation was of very low safety significance and was entered into Entergy's corrective action program (CR-PNP-2011-1377), this violation is being treated as an NCV, consistent with the NRC's Enforcement Policy. **(NCV 05000293/2011003-03, Inadequate Risk Assessment for Planned Maintenance and Testing on RCIC, SLC and ATS Systems)**.

1R15 Operability Evaluations (71111.15)

.1 Quarterly Review of Operability Evaluations

a. Inspection Scope (8 samples)

The inspectors reviewed eight 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 UFSAR requirements. The documents reviewed during this inspection are listed in the Attachment. The inspectors reviewed the following degraded or non-conforming conditions:

- CR-PNP-2011-1594, Intermediate Range Monitor / Average Power Range Monitor Overlap Data Outside of Procedural Tolerance;
- CR-PNP-2011-2274, While Performing Bus B6 Transfer System Relay Calibrations, the Time Delay Contacts Operated Instantaneously;
- CR-PNP-2011-2024, Post Work Test Results for the Start-up Transformer Did Not Meet All Acceptance Criteria;
- CR-PNP-2011-2188, Acceptance Criteria for 'B' Emergency Diesel Generator Start and Close on Bus A6 Within 10.6 Seconds Was Not Met;
- CR-PNP-2011-2556, Error Identified by General Electric in the Pilgrim Loss of Coolant Accident Analysis Affecting Peak Centerline Temperature;
- CR-PNP-2011-2635, Safety Relief Valve 'C' Operability After Forced Outage;
- CR-PNP-2011-3007, Recirculation Pump Flow Converter Failure Alarms; and
- CR-PNP-2011-3021/3049, Control Room Vital Area Door Inoperable.

b. Findings

Introduction: The inspectors identified a Green NCV of Technical Specification (TS) 3.7.B.2.f, "Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (CRHEAFS)," for Entergy's failure to enter and perform the actions prescribed in TS after the Control Room Envelope (CRE) was breached during work on a vital area door leading into the CRE. Entergy has since repaired the vital area door and restored the CRE.

Description: On June 20, 2011, the inspectors identified that a vital area door into the control room was open, and that a security officer had been stationed as a security compensatory measure. The inspectors noted that the open door would also affect the integrity of the CRE. The inspectors discussed the open door and its impact on the CRE with Operations to determine what actions were being taken to mitigate the cause of the inoperable CRE, as required by TS 3.7.B.2.f. However, Operations was not aware of the work being performed on the vital area door, and therefore, had not initiated mitigative actions; did not ensure that the mitigating actions would be effective to ensure occupant exposures to radiological, chemical, and smoke hazards would not exceed limits; and did not ensure the CRE boundary would be restored within 90 days, as required by TS 3.7.B.2.f. The CRE vital area door was repaired and the CRE restored later that day.

Analysis: The inspectors determined that Entergy not entering and performing the actions required by TS 3.7.B.2.f was a performance deficiency. This condition did not impact the regulatory process and did not contribute to any actual consequences, therefore Traditional Enforcement does not apply. The inspectors determined that the issue was more than minor because the finding was associated with the Human Performance attribute of the Barrier Integrity cornerstone (maintain the radiological barrier function of the control room), and adversely affected the cornerstone's objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the inoperable CRE could affect the operator's ability to occupy the control room under adverse radiological, chemical or smoke conditions while responding to an event. IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1- Initial Screening and Characterization of Findings," was used to evaluate the impact of the finding on the loss of operability or functionality of the CRE and CHREAFS. The inspectors determined that further evaluation was required since the finding had the potential to impact the control room envelope due to the effects of smoke and toxic gas. As a result of this screening, a Phase 3 evaluation was conducted by a Senior Reactor Analyst (SRA). The SRA conducted a qualitative evaluation and determined the risk impact on control room habitability from smoke and toxic gas to be low (Green). Specifically, the Pilgrim Station Individual Plant Examination for External Events (IPEEE), sections 5.3.3 and 5.3.4, identified that the overall risk from on-site and off-site chemical release was low. Smoke impacts on the control room were evaluated and determined to be low for the identified exposure period. This was mainly due to the fact that the risk significant impacts from smoke would be from fires and smoke originating within the control room envelope and the impairment on the barrier would not impact this state.

The inspectors determined that this issue had a cross-cutting aspect in the Work Control component of the Human Performance cross-cutting area. Specifically, Entergy did not plan and coordinate work activities affecting the CRE such that interdepartmental coordination assured plant and human performance. In this case, Operations was not made aware that Maintenance would be working on the control room vital door [H.3(b)].

Enforcement: TS 3.7.B.2.f, "Standby Gas Treatment System and Control Room High Efficiency Air System (CHREAFS)," requires that immediate actions be taken to mitigate the cause of an inoperable CRE, that within 24 hours, the effectiveness of these actions will ensure CRE occupants do not exceed radiological, chemical, or smoke hazard limits, and that the CRE is restored to operable status within 90 days. Contrary to the above, on June 20, 2011, Entergy did not take immediate mitigative actions or evaluate the effectiveness of actions to ensure limits would not be exceeded. Corrective actions included entering the TS, repairing the vital area door, and initiating an apparent cause review. Because the violation was of very low safety significance and Entergy has entered it into their corrective action program (CR-PNP-2011-3021 and -3049), this violation is being treated as an NCV consistent with the NRC's Enforcement Policy. **(NCV 05000293/2011003-04, Failure to Enter Technical Specifications for CHREAFS).**

.2 Resolution of Unresolved Item (URI) 05000293/2011002-01 "Application of Technical Specification (TS) 3.3.B.1 when Control Rod Position Indication is Lost."

a. Inspection Scope

URI 05000293/2011002-01, "Application of TS 3.3.B.1 when Control Rod Position Indication is Lost," was opened to determine Entergy's position on whether TS 3.3.B.1, Control Rod Operability, should have been entered when the Rod Position Indication System failed on January 20, 2011. Entergy reviewed their TS requirements, discussed the issue with their peers, and identified that the operability of control rods was affected when control rod position indication was lost. The inspectors reviewed Entergy's response, applicable Technical Specifications, and UFSAR requirements. The documents reviewed are listed in the Attachment.

b. Findings

Introduction: The inspectors identified a Green NCV of Technical Specification (TS) 3.3.B.1, "Control Rod Operability," for Entergy's failure to enter and perform the actions prescribed in TS after losing control rod position indication. Entergy has since restored control rod position indication by repairing a failed power supply.

Description: On January 20, 2011, at 5:19 PM, control room personnel at Pilgrim Nuclear Power Station lost position indication of the control rods. Instrumentation and control (I&C) technicians began troubleshooting the Rod Position Indication System (RPIS) and identified that a power supply had a damaged cooling fan. Operations notified the resident inspector staff and noted that TS surveillance 4.3.B.1.5, "Control Rod Operability," had been completed successfully less than an hour prior to losing RPIS. TS 3.3.B.1 was not entered at the time of discovery. Operations determined that TS entry was not required until the 24 hour periodicity of surveillance 4.3.B.1.5 expired. However, the inspectors determined that Pilgrim's TS Bases describe the factors in determining the operability of a control rod:

"The OPERABILITY of an individual control rod is based on a combination of factors, primarily the scram insertion times, the associated control rod scram accumulator status, the control rod coupling integrity, and the ability to determine control rod position."

According to TS 3.3.B.1, all control rods without position indication were required to be declared inoperable and fully inserted into the core within 3 hours. In addition, the associated Control Rod Drive for each control rod was required to be disarmed within 4 hours.

I&C technicians replaced the damaged cooling fan to repair the power supply and RPIS was restored at 9:53 PM. Control room personnel observed that there had been no change with respect to control rod position. Condition Report CR-PNP-2011-0272 was written to address the power supply failure and Condition Report CR-PNP-2011-0511 was subsequently written to address Entergy's administration of TS 3.3.B.1.

Analysis: The inspectors determined that Entergy's failure to enter and perform the actions required by TS 3.3.B.1 was a performance deficiency. This condition did not impact the regulatory process and did not contribute to any actual consequences; therefore, Traditional Enforcement did not apply. The inspectors determined that the issue 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 reliability of systems that respond to events to prevent undesirable consequences (i.e., core damage). Specifically, the locations of the control rods were indeterminate, which could substantially impact operator's abilities to implement Emergency Operating Procedures. IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1-Initial Screening and Characterization of Findings," was used to evaluate the significance of the finding. Attachment 0609.04, Table 4a, was used to evaluate the impact of the finding on loss of operability or functionality. The inspectors determined that the function of the control rods to add negative reactivity to the core during an event was not affected (SCRAM time and control rod worth were not affected). In addition, alternate means were available to operators to determine control rod position and once the power supply was restored, the control rods were determined to have remained in their original positions. Also, since the finding is not potentially risk significant due to a seismic, flooding or severe weather initiating event, the finding was determined to be of very low safety significance (Green).

The inspectors determined that this issue had a cross-cutting aspect in the Decision Making component of the Human Performance cross-cutting area. Specifically, Entergy did not use conservative assumptions in decision making and adopt a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disprove the action. In this case, Entergy did not take the conservative approach to enter TS 3.3.B.1 when faced with a degraded condition affecting control rod position indication [H.1(b)].

Enforcement: TS 3.3.B.1 "Control Rod Operability," requires that inoperable controls rods be fully inserted into the core within 3 hours when control rods become inoperable. Contrary to the above, on January 20, 2011, Entergy did not enter TS 3.3.B.1 following the loss of control rod position indication which would have required the rods to be declared inoperable. Corrective actions included replacing the power supply and restoring control rod position indication. Because this violation was of very low safety significance and Entergy has entered it into their corrective action program (CR-PNP-2011-0511 and CR-PNP-2011-0272), this violation is being treated as an NCV, consistent with the NRC's Enforcement Policy. **(NCV 05000293/2011003-05, Failure to Enter Technical Specifications after Loss of Control Rod Position Indication)**

1R18 Plant Modifications (71111.18)

.1 Temporary Modification for Reactor Shutdown / Flood-up Level Indication

a. Inspection Scope (1 sample)

The inspectors reviewed procedural temporary modification 3.M.2-40, "Refuel Outage Temporary Modification Reactor Shutdown / Flood-up Level Indication," to determine whether reactor vessel level indication would be maintained by the modification. The inspectors reviewed Control Room drawings, work orders, and procedures to ensure the temporary modification would adequately reflect reactor vessel level and that no other instrumentation in the field was affected by the modification. The inspectors reviewed the annotated drawings to determine whether they properly reflected the temporary modification. The inspectors also walked down the Control Room and the level instrumentation in the reactor building to review the installed modification. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2 Permanent Modification for Increasing the Safety Relief Valve (SRV)/Spring Safety Valve (SSV) Setpoints and Tolerances and SRV/SSV Replacement

a. Inspection Scope (1 sample)

The inspectors reviewed permanent modification EC 5000071989, Revision 7, "SRV/SSV Setpoint and Tolerance Increase and Replacement," and the associated 10 CFR 50.59 screening, to determine whether the licensing bases and performance capability of the SRV/SSV systems had been degraded through the modification. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.3 Permanent Modification of 'A' EDG for Replacement of Governor Control System

a. Inspection Scope (1 sample)

The inspectors reviewed permanent modification EC 0000005974, Rev 0, "Upgrade EDG "A" Governor Control System," and the associated 10 CFR 50.59 screening, to determine whether the licensing bases and performance capability of the 'A' EDG had been degraded through the modification. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (6 samples)

The inspectors reviewed six samples of post-maintenance tests during this inspection period. The inspectors reviewed these activities to determine whether the post-maintenance test 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:

- Governor modification on the 'A' Emergency Diesel Generator;
- Replace Reactor Recirculation Sample Valve;
- Replace Safety Relief Valves;
- 'A' Control Rod Drive Replacement;
- 'A' Recirculation Pump Motor Generator Set Maintenance; and
- Replace Main Steam Line 'B' Outboard Isolation Valve Actuator.

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20)

.1 Refueling Outage 18

a. Inspection Scope (1 sample)

Periodic review of RFO 18 Work Plan and Outage Risk

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

Monitoring of Shutdown Activities

The inspectors observed operators performing portions of the reactor shutdown, and plant cooldown to assess operator performance with respect to communications, command and control, procedure adherence, and compliance with Technical Specification cooldown limits. Upon shutdown, the inspectors also performed an inspection walkdown of the drywell to verify the integrity of structures, piping and supports, and to confirm systems appeared functional.

Clearance Activities

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

Reactor Coolant System (RCS) Instrumentation

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

Electrical Power

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

RHR and SFP System Monitoring

The inspectors observed the RHR and SFP system status and operating parameters to verify that the cooling systems operated properly. Verification included periodic review of the SFP and reactor cavity level, temperature, and RHR system flow. A complete system walk down was completed for the SFP Cooling System.

Inventory Control

The inspectors reviewed Entergy's actions to establish, monitor, and maintain the proper water inventory in the reactor vessel and spent fuel pool. The inspectors reviewed the plant system flow paths and configurations established for reactor makeup and verified the configurations were consistent with the outage plan.

Foreign Material Exclusion (FME)

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

Control of Heavy Loads

The inspectors reviewed licensee actions to control the lift of heavy loads during the outage. The inspectors reviewed licensee actions to manage the increased risk during these activities and to implement compensatory measures to protect the integrity of systems important to safe shutdown.

Containment Control

The inspectors reviewed licensee activities during the outage to control primary and secondary containment and to clean and prepare the containment for closure prior to plant restart. The inspectors performed periodic tours of the drywell to review the control of work activities and containment conditions. The inspectors performed a walkdown of the drywell prior to reactor startup to review licensee cleanup and demobilization controls in areas where work was completed to assure that tools, materials and debris were removed.

Fuel Shuffle Activities and Reactivity Control

The inspectors verified that refueling activities were performed in accordance with core alterations Technical Specifications, including the requirements for core monitoring using the source range monitors and the functional checks of the refueling interlocks. The inspectors observed communications and the coordination of activities between the control room, the General Electric physicist, and the refueling floor while fuel handling activities were in progress.

Monitoring Heatup and Startup Activities

The inspectors observed and/or reviewed heatup and startup activities during the period of May 10, 2011 through May 14, 2011. The inspection consisted of control room observations, plant walkdowns, and a review of control board indicators, operator logs, plant computer information, and station procedures. The inspectors observed operator actions including the preparations for the approach to criticality, reactor critical operations, low power operations, and the synchronization of the main turbine generator to the electrical grid. The inspectors observed plant restart and power ascension to verify that Technical Specifications, license conditions, and other requirements for mode changes were met. The inspectors responded to the control room following a reactor

SCRAM at low power. A special inspection team was chartered to review the circumstances surrounding the SCRAM and the licensee response.

Problem Identification and Resolution

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

b. Findings

A Special Inspection Team (SIT) was chartered to review the circumstances surrounding the reactor scram event on May 10, 2011, and the subsequent start-up activities. A separate inspection report (05000293/2011012) will be issued to document this review and the results of the SIT.

.2 Forced Outage 19-1

a. Inspection Scope

The inspectors reviewed the outage plan and shutdown risk assessments for a forced, non-refueling outage performed from May 14, 2011, through May 19, 2011. The outage was performed following a plant shutdown due to the inability to meet Technical Specification Drywell to Torus differential pressure requirements. The documents reviewed during the inspection are listed in the Attachment. During this outage, the inspectors observed plant shutdown and start-up activities including the outage activities listed below:

- Hot and Cold Shutdown Cooling Control;
- Shutdown Risk Assessment and Risk Management;
- Implementation of Technical Specifications;
- Outage Control Center Activities;
- Plant Startup; and
- Licensee identification and resolution of problems identified, during, and related to outage activities.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (7 samples)

The inspectors witnessed seven 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 documents reviewed during this inspection are listed in the Attachment. The following surveillance tests were evaluated:

- Manual Start and Loading of the Station Blackout Diesel Generator via Safety Bus A5/A6;
- Secondary Containment Leak Rate Test;
- Main Steam Isolation Valve Inboard and Outboard Local Leak Rate Tests (CIV);
- Loss of Offsite Power Load Testing of the Emergency Diesel Generators;
- High Pressure Coolant Injection operability run from the Alternate Shutdown Panel;
- Analog Trip System Logic Testing for Emergency Core Cooling Systems; and
- Reactor Coolant System Leakage (RCS).

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 sample)

The inspectors observed a licensed operator requalification training simulator exercise on June 9, 2011. The inspectors evaluated operator performance in the simulator for a loss of Turbine Building Closed Cooling Water and a High Pressure Coolant Injection steam leak which resulted in the need for emergency depressurization and escalated to an Alert. The inspectors assessed the implementation of Emergency Action Level classification and notification decisions for this event. The inspectors also assessed whether Pilgrim's critique of the exercise assessed all observations and findings.

b. Findings

No findings were identified.

2. RADIATION SAFETY (RS)

Cornerstone: Occupational and Public Radiation Safety

2RS01 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope (1 sample)

During the period of April 25, 2011 through April 28, 2011, the inspectors performed the following activities to verify that Entergy properly assessed the radiological hazards in the workplace and implemented appropriate radiation monitoring and exposure controls during refueling outage operations. Implementation of these controls was reviewed against the criteria contained in 10 CFR Part 20, relevant Technical Specifications, and the licensee's procedures.

Inspection Planning

- The inspectors reviewed radiation protection program self assessments and audits.

Radiological Hazard Assessment

- The inspectors walked down the facility, including the reactor building, the drywell, radwaste processing, the turbine building, and the condenser bay to evaluate material and radiological conditions. The inspectors verified the integrity and postings of the Locked High Radiation Areas (LHRA) in the reactor building and one Very High Radiation Area (VHRA).
- The inspectors reviewed pre-work and in-progress surveys for the reactor disassembly.
- The inspectors verified for five (5) air samples that they were collected and analyzed in accordance with licensee procedures.

Instructions to Workers

- The inspectors verified that Entergy had established a means to inform workers of changes that could significantly impact their occupational dose. The licensee has a central monitoring system, alarming electronic dosimeters with transmitting capability, and head set radios for each job in the drywell.

Contamination and Radioactive Material Control

- The inspectors reviewed Entergy's procedure for the survey and release of material and verified it is sufficient to control the spread of contamination and prevent the unintended release of radioactive materials from the site.
- The inspectors observed the surveys of material at the Radiologically Controlled Area (RCA) exit point and the actions taken when alarms occurred. The inspectors verified that the surveys and actions taken in response to alarms were in accordance with the licensee's procedures.

Radiological Hazards Control and Work Coverage

- The inspectors verified the placement of radiation monitoring devices on selected individuals.
- The inspectors reviewed Radiation Work Permits (RWP) for In Service Inspection (ISI) and valve work inside the drywell. The inspectors reviewed a Total Effective Dose Equivalent (TEDE) evaluation for one task.

b. Findings

No findings were identified.

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope (1 sample)

During the period of April 25, 2011 through April 28, 2011, the inspectors performed the following activities to verify that Entergy was properly implementing operational, engineering, and administrative controls to maintain personnel exposure as low as reasonably achievable (ALARA). Implementation of these controls was reviewed against the criteria contained in 10 CFR Part 20, applicable industry standards, and the licensee's procedures.

Verification of Dose Estimates and Exposure Tracking Systems

- The inspectors reviewed the assumptions and basis described in the RWP and ALARA packages for ISI activities, reactor disassembly and reassembly activities, scaffold activities, shielding activities, and main steam relief valve and safety valve replacements.
- The inspectors verified for the above activities that the licensee had established measures to track, trend, and adjust occupational dose estimates for ongoing work activities. The inspectors verified trigger points were used to prompt additional reviews. The inspectors reviewed Entergy's method for adjusting exposure estimates when unexpected changes in scope, dose rates, or emergent work are encountered.

b. Findings

No findings were identified.

2RS03 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

a. Inspection Scope (1 sample)

During the period of April 25, 2011 through April 28, 2011, the inspectors performed the following activities to verify that Entergy was controlling in-plant airborne concentrations consistent with ALARA. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and Entergy's procedures.

Engineering Controls

- The inspectors verified that Entergy used ventilation systems as part of its engineering controls to control airborne radioactivity.
- The inspectors verified the blast tent on the turbine 51' elevation and the turbine valve work ventilation unit efficiencies and airflow capacities are consistent with maintaining concentrations of airborne radioactivity in work areas below the concentrations of an airborne area to the extent practicable and are consistent with Entergy's procedural guidance and ALARA.
- The inspectors verified the reactor building vent monitoring system has an alarm and set-point that are sufficient to prompt Entergy and workers to take action to

ensure that doses are maintained within the limits of 10 CFR Part 20 and ALARA.

- The inspectors verified that Entergy had established trigger points for evaluating levels of airborne beta-emitting and alpha-emitting radionuclides.

Use of Respiratory Protection Devices

- The inspectors verified that Entergy provided respiratory protective devices such that occupational doses are ALARA. The inspectors verified that Entergy performed an evaluation concluding that the use of respirators is ALARA for the turbine grit blast project. The inspectors also verified that the level of protection provided by the respiratory protection devices during use is consistent with assumptions used in the Entergy's work controls and dose assessment.
- The inspectors verified respiratory protection devices used were National Institute for Occupational Safety and Health (NIOSH) certified.
- The inspectors verified that three individuals working in the grit blast tent were qualified to wear respiratory protection equipment by reviewing applicable training records and physical examination records.

b. Findings

No findings were identified.

2RS04 Occupational Dose Assessment (71124.04)

a. Inspection Scope (1 sample)

During the period of April 25, 2011 through April 28, 2011, the inspectors performed the following activities to verify that Entergy appropriately monitors occupational dose. Implementation of these controls was reviewed against the criteria contained in 10 CFR Part 20, applicable industry standards, and the licensee's procedures.

Special Dosimetric Situations

- The inspectors verified there were no individuals who declared pregnancy. The inspectors verified that the licensee's radiological monitoring program was technically adequate to assess dose to an embryo/fetus.
- The inspectors reviewed Entergy's methodology for monitoring external dose in situations in which non-uniform fields are expected. The inspectors verified that Entergy had established criteria for determining when alternate monitoring techniques are to be used.
- The inspectors reviewed dose assessments performed for the ISI core spray nozzle inspection work where multiple badges were worn. The inspectors verified that the assessments were performed consistently with the licensee's procedures and dosimetric standards.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator (PI) Verification (71151)

.1 Cornerstone: Initiating Events

a. Inspection Scope (3 samples)

The inspectors reviewed Performance Indicator (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, Condition Reports (CRs), Licensee Event Reports (LERs), 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." The documents reviewed during the inspection are listed in the Attachment. The following performance indicators were reviewed:

- Unplanned SCRAMs per 7000 Critical Hours;
- Unplanned SCRAMs with Complications; and
- Unplanned Power Changes per 7000 Critical Hours.

b. Findings

No findings were identified.

.2 Cornerstone: Occupational/Public Radiation Safety

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 condition 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, "Regulatory Assessment Performance Indicator Guideline," 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 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 corrective action program. 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 were identified.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope (1 Sample)

The inspectors performed a review of Entergy's 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 January 2011 through June 2011, although the inspectors also evaluated previous trend results for CRs and observations from selected inspection samples from June 2010 through December 2010. The documents reviewed during the inspection are listed in the Attachment.

b. Findings and Observations

No findings were identified. One trend was identified and is discussed below.

Implementation of the Operability Determination Process

The inspectors have observed deficiencies in the areas of Operability Determination quality, timeliness, conservative decision making, and entry into Technical Specifications. The inspectors discussed these observations at the time of their occurrence, during quarterly exit meetings, and during semi-annual trend review discussions. Training has been conducted by the Operations and Engineering departments and improvements were subsequently noted in the quality and level of detail in some operability samples. In addition, CR-PNP-2011-0137 and CR-PNP-2011-1140 were written by the Operations department in January and March of 2011 respectively, to assess operability shortfalls and to address programmatic areas for improvement. As a corrective action to CR-PNP-2011-0137, further operability training was conducted by the Operations Department. However, additional recent examples have been identified by the inspectors during the past six months, including:

- A non-cited violation was identified for not entering and performing the actions in Technical Specification 3.3.B.1 "Control Rod Operability", after Entergy lost control rod position indication.

- An operability evaluation of a degraded "B" Recirculation Flow Comparator didn't address the degraded condition's impact on the operability of the Average Power Range Monitor scram setpoint at lower power levels.

- An operability evaluation of a faulty B6 Transfer System time delay relay did not address the purpose of the time delay or the acceptability of the degraded condition.

- Leakage into the "B" RBCCW heat exchanger from salt service water was identified, however, an evaluation of the leakage and a conclusion of inoperability were not completed for over 7 hours. In addition, Operations had not established a reasonable assurance of continued operability in the intervening timeframe.

The inspectors have concluded that these operability determination issues constitute a trend with the implementation of this program. The inspectors will follow Entergy's corrective actions per CR-PNP-2011-0137 to evaluate their response to this trend.

.3 <u>Annual Sample: Review of Refueling Outage-17 (RFO-17) Drywell Inspection Corrective Actions</u>

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

The inspectors selected Condition Report (CR)-PNP-2009-2408, which documented the results of the NRC inspection of the drywell prior to start-up from RFO-17, for a detailed follow-up review. The NRC inspection had identified a significant amount of equipment, tools and debris in the drywell following the completion of drywell cleanup and inspections by the licensee. In addition, the inspectors noted that no specific activity for drywell clean-up was identified in the schedule. Also, little guidance was provided to maintenance supervisors who were assigned to conduct the inspections to support drywell closeout.

The inspectors assessed Entergy's cause analysis, extent of condition review, the prioritization and timeliness of corrective actions, and whether the planned or completed corrective actions were appropriate to prevent recurrence. Additionally, the inspectors performed a drywell walkdown and inspection to determine the effectiveness of these corrective actions. Also, the inspectors performed interviews with cognizant plant personnel in Outage Management, Scheduling, and the Maintenance and Operations departments.

b. <u>Findings and Observations</u>

No findings were identified.

The inspectors determined that Entergy's corrective actions were effective in improving the clean-up and inspection of the drywell following a refueling outage. The inspectors continued to identify tools and debris during the NRC RFO-18 drywell inspection; however, the amount of material identified was significantly less than that identified during RFO-17. The inspectors noted that reinforcement of the "clean as you work" philosophy and a planned clean-up of legacy debris in the drywell should improve the conditions in the drywell during and following future outages. The inspectors also identified that some corrective actions associated with the apparent and contributing causes of CR-PNP-2009-2408 were closed to a Learning Organization (LO)-Outage

Lessons Learned (OLP)-2009-0006 Condition Report. This is not in accordance with Entergy procedure, EN-LI-102, "Corrective Action Process." In addition, some of these corrective actions were subsequently closed with no action taken. The decision to close these specific corrective actions was not reviewed by the owner of CR-PNP-2009-2408.

The inspectors determined that the closure of the corrective actions contrary to guidance contained in Entergy procedure EN-LI-102 represented a performance deficiency. The performance deficiency was not more than minor because it was not a precursor to a significant event, it would not lead to a more significant safety concern if left uncorrected and it did not adversely affect any of the ROP cornerstone objectives. In accordance with NRC Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," the above issue constitutes an issue of minor significance that is not subject to enforcement action in accordance with the NRC's Enforcement Policy.

.4 Identification and Resolution of Problems- In-service Inspection

a. Inspection Scope

The inspector reviewed a sample of CRs initiated during ISI examinations this outage and placed in the corrective action process for evaluation and disposition. Also, CR-PNP-2009-1514 from the previous outage (RFO 17) was reviewed by the inspector for comparison with the results of NDE performed this outage (RFO 18) to determine if any change had occurred during this operating cycle. The inspector reviewed CR-PNP-2009-1514 (linear indication at RPV head to flange weld) and the analytical evaluation performed. The analysis supported the condition disposition as meeting all design bases requirements and the component is fully functional with no further action required. The Magnetic Particle Test (MT) performed during this outage (Report RFO 18-005) confirmed there was no change in the indication size and characteristics.

The inspector also reviewed CR-PNP-2011-2210 which was initiated as a result of the ASME Section XI, IWE visual inspection of the protective coating on accessible interior surfaces of the torus and exposed surfaces of the primary containment liner, and identified several areas of coating deterioration, minor rusting with some peeling, and flaking of coating (primarily within the torus). The inspector verified that the conditions identified were entered into the licensee's corrective action program for engineering evaluation and disposition.

b. Findings

No findings were identified.

4OA3 Event Follow-up (71153)

.1 Operator Performance During Momentary Loss of Instrument Power Bus Y1

a. Inspection Scope (1 sample)

The inspectors observed an unplanned momentary loss of instrument power to bus Y1 on April 13, 2011. The control room received multiple alarms, and operators completed the actions of Procedure 5.3.7, "Loss of Instrument Power Bus Y1." The inspectors reviewed procedural guidance for the loss of instrument power to bus Y1 and observed control room conduct and control of the event.

b. Findings

No findings were identified.

.2 Reactor Scram Due to High Intermediate Range Monitor (IRM) Reactor Protection System (RPS) Channel Trips

a. Inspection Scope (1 sample)

On May 10, 2011, the Pilgrim reactor scrammed due to High IRM RPS channel trips. Operators responded to the plant scram and plant systems responded normally. Initial review of the scram identified the likely causes as IRM malfunctions or unanticipated high notch rod worth of the control rod being withdrawn at the time. The inspectors responded to the control room and reviewed the operators' response to the scram. At 0200 on May 11, 2011, Entergy notified the inspectors that they had determined that the cause of the scram was due to operator error. The inspectors responded to the site to evaluate Entergy's post trip report and decision to start up the reactor. The inspectors reviewed the circumstances of the scram, corrective actions, and the subsequent decision to start-up with NRC management. The decision was made to conduct follow-up inspection of the event and to charter a Special Inspection Team (SIT). The SIT arrived on-site on May 16, 2011. Following a review of the Entergy root cause analysis, a SIT report will be issued to document any findings associated with their review of this event.

b. Findings

A separate SIT report (05000293/2011012) will be issued to document any findings associated with the review of this event.

.3 'A' and 'B' Containment Gaseous and Particulate Monitoring Channels Declared Inoperable

a. Inspection Scope (1 sample)

On June 27, 2011, Entergy identified that valve CV-5065-92 failed in the closed position. With CV-5065-92 failed closed, both 'A' and 'B' trains of the drywell gaseous and particulate radioactivity monitoring channels were declared inoperable. Entergy entered the appropriate Technical Specifications which allowed continued reactor operation for up to 30 days provided drywell atmosphere grab samples were analyzed every 12 hours. A grab sample was taken within 12 hours on June 28, 2011 and analyzed to be within the expected range for drywell radioactivity. The inspectors reviewed Entergy's actions, Technical Specifications, and procedures.

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)."

a. Inspection Scope

On May 14, 2011, the inspectors completed a review of Entergy's severe accident management guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990's, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee's training of personnel on the use of SAMGs, and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for Pilgrim Nuclear Power Station were provided in an Attachment to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated May 27, 2011 (ML111470361).

b. Findings

No findings were identified.

.2 (Closed) Unresolved Item (URI) 05000293/2011002-03; Need For Clarification on Condensate Storage Tank Suction Piping ASME Classification

a. Inspection Scope

A Problem Identification and Resolution (PI&R) sample inspection was conducted during the period January 3, 2011 through January 12, 2011. As a result of this review, a URI was identified. The inspector noted a discrepancy between various Piping and Instrumentation Drawings (P&ID) for the condensate and demineralized water storage/transfer systems and the associated In-service Inspection (ISI) drawings for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) piping and their related ASME Code safety classification. Condition Report (CR) PNP-2010-2645 addresses this discrepancy and a follow up inspection was conducted to evaluate additional information provided in the disposition. This disposition provided supplementary information and clarification of the PNPS process that was used in the assignment of the original piping classifications (safety classification) and provides guidance for the current classification assignment. The inspector reviewed the supplementary information provided in the Evaluation Response to Corrective Action #18. Recommendations were provided for incorporation in Corrective Action #21 which specifies the preparation of an Engineering Change that will update the applicable portions of FSAR Appendix "A" and Specifications M300, M301, M305, and M605.

The inspector noted that the historical presentation of the piping classification process at Pilgrim Station evolved into Class 1 Piping being different from ISI Safety Class 1, 2 and

3, per ASME Code. Today, Class 1 Piping reflects the use of certain codes/standards that are different then the ASME Code Classifications. The CR addressed the need to update the FSAR Appendix "A" and the referenced specifications and applicable piping and instrument diagrams. The subject piping for the condensate storage tank is not ASME Safety Class 1, 2 or 3, but is appropriately classified as Class 1 piping. Also, the licensee representative indicated intent to clarify the symbol legend on applicable drawings to avoid user mis-interpretation of the drawing. The preparation and issue of Corrective Action #21 (Engineering Change) is intended to capture the update requirements of the CA#18 evaluation which includes FSAR Appendix "A" and related specifications M300, M301, M305 and M605.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On April 28, 2011, a Radiation Safety exit meeting was performed with Mr. Robert Smith and other members of the Pilgrim staff. The inspectors confirmed that no proprietary information was provided during the inspection.

On May 6, 2011, an In-service Inspection exit meeting was performed with Mr. Martin Mantenfel and other members of the Pilgrim Staff. The inspectors confirmed that no proprietary information was provided during the inspection.

On July 14, 2011, the resident inspectors performed an exit meeting and presented the preliminary inspection results to Mr. Robert Smith, 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy personnel:

S. Bethay	Director, Nuclear Safety Assurance
D. Brugman	Radiation Protection Supervisor
B. Chenard	System Engineering Manager
J. Dreyfuss	Plant General Manager
V. Fallacara	Engineering Director
W. Lobo	Licensing Engineer
J. Lynch	Licensing Manager
J. Macdonald	Assistant Operations Manager-Shift
T. McElhinney	Chemistry Manager
D. Noyes	Operations Manager
J. Priest	Radiation Protection Manager
J. Scheffer	Chemistry Supervisor
K. Sejkora	Staff Chemist
R. Smith	Site Vice President
J. Taormina	Maintenance Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

NCV 05000293/2011003-01	Transient Combustible Loading in SLC Room in Excess of the Fire Hazards Analysis Limit (Section 1R05)
FIN 05000293/2011003-02	Submerged Medium Voltage Cables (Section 1R06)
NCV 05000293/2011003-03	Inadequate Risk Assessment for Planned Maintenance and Testing on RCIC, SLC and ATS Systems (Section 1R13)
NCV 05000293/2011003-04	Failure to Enter Technical Specifications for CHREAFS (Section 1R15)
NCV 05000293/2011003-05	Failure to Enter Technical Specifications after Loss of Control Rod Indication (Section 1R15)

Closed

URI 05000293/2011002-01	Application of TS 3.3.B.1 When Control Rod Position Indication is Lost (Section 1R15)

URI 05000293/2011002-03 Need For Clarification on Condensate Storage Tank Suction Piping ASME Classification (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

Section 1R01

Procedure 5.2.2, Revision 31, High Winds (Hurricane)
Procedure 2.1.37, Revision 28, Coastal Storm Preparation Actions
Procedure 8.C.40, Revision 25, Season Weather Surveillance
Procedure 1.4.4, Revision 21, New England Power Grid Operations/Interfaces
Procedure 1.5.22, Revision 12, Risk Assessment Process
Procedure 2.1.14, Revision 105, Station Power Changes
ISO New England Master LLC procedure #1, Nuclear Plant Transmission Operations
REMVEC Operating Procedure #3, Scheduling Outages of REMVEC Transmission Facilities
Final Safety Analysis Report (FSAR), Section 8.3, Standby AC Power Source
NRC GL-2006-02, Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power
Pilgrim's Responses to GL-2006-002
FSAR, Section 8.10, Blackout AC Power Source
FSAR, Section 2.4.4, Storm Flooding Protection
Technical Specifications (TS) 3.5.F, Minimum Low Pressure Cooling and Diesel Generator

Section 1R04

Procedure 2.2.8, Revision 96, Standby AC Power System
Procedure 2.2.85.1, Revision 15, Augmented Fuel Pool Cooling (with Shutdown Cooling) Mode 1
Procedure 2.2.85, Revision 77, Fuel Pool Cooling and Filtering System
Procedure EN-DC-205, Revision 2, Functional Failure Determination Form
Procedure 2.2.24, Revision 46, Standby Liquid Control System
Procedure 2.2.146, Revision 42, Station Blackout Diesel Generator
Work Order (WO)# 00346836, Addressed in EN-DC-205, Functional failure determination form attached
WO#00219084, Addressed in EN-DC-205 attached
CR-PNP-2010-2450, Received "Refueling Bellows Seal Rupture" alarm associated with FS-4803.
CR-PNP-2010-3650, Received Fuel Pool Cooling Panel alarm for Refueling Bellows Failure
CR-PNP-2010-4084, While performing the weekly swap of fuel pool cooling pumps chattering noise was noticed coming from the discharge check valve of fuel pool cooling pump P-201A
CR-PNP-2010-4081, Engineering Response Memo (ERM) #91-220 approved the continued use of Teflon tape on HCU valves and fittings based on GE sil#128
CR-PNP-2010-3418, Upon investigation, no abnormal conditions were found with the running or standby pumps. Alarm setpoint is 40 psid
CR-PNP-2010-2745, FPC Panel Alarm Received
CR-PNP-2011-0492, FI-4868, 4869, 4870 tolerances too low
CR-PNP-2011-1384, Isolation of fuel pool demin during resin removal unable due to 19-HO-132 leak-by.
CR-PNP-2011-2066, During ISI inspection of the IWE-LINDERDRAINS on 74' RB, Drain line

was leaking approximately 100 drops per minute
Final Safety Analysis Report (FSAR), Section 10.4, Revision 23, Fuel Pool Cooling and Cleanup System
FSAR, Section 8.10, Blackout AC Power Source
System Health Report FPC
Maintenance Rule Basis Document
M264 P&ID Drawing, Station Blackout, Revision 18
Technical Specification 3.5.F.1, Minimum Low Pressure Cooling and Diesel Generator Activity

Section 1R05

Procedure 5.5.2, Revision 46, Special Fire Procedure
Procedure 8.B.14, Revision 44, Fire Protection Technical Requirements
Procedure 8.B.17.1, Revision 20, Inspection of Fire Door Assemblies
Procedure EN-DC-161, Revision 4, Control of Transient Combustibles
CR-PNP-2011-2149, No TCE initiated for Transient Combustibles on 91'
CR-PNP-2011-2214, 5 Portable Fire Extinguishers in 'A' Switchgear room had not received their monthly inspections
EN-DC-161, Revision 4, Control of Transient Combustibles
Fire Hazards Analysis Fire Area 1.9, Fire Zone 2.2, 'A' Switchgear and Load Center Room
Fire Hazards Analysis Fire Area 1.10, Fire Zone 4.1, 'B' Emergency Diesel Generator Room
Fire Hazards Analysis Fire Area 1.10, Fire Zone 1.23, Standby Gas Treatment System Room
Fire Hazards Analysis Fire Area 1.9, Fire Zone 1.15, Standby Liquid Control System Room
Fire Hazards Analysis Fire Area 1.30, Fire Zone 1.30, Drywell
Transient Combustible Evaluation 11-033

Section 1R06

NRC Information Notice 2010-26, Submerged Electrical Cables
Work Order#52311755 01, Inspect Appendix R, Manhole #26B (RHR, Core Spray, EDG Cables)
Work Order#52345427 01, Inspect Manhole #2A (Start-up Transformer Cables)
Work Order#52345426 01, Inspect Manhole K (Station Blackout Diesel Cables)
NUREG/CR-7000, BNL-NUREG-90318-2009, Essential Elements of an Electric Cable Condition Monitoring Program
Procedure EN-DC-346, Revision 1, Cable Reliability Program
CR-PNP-2011-2911, Partially Submerged Cables Identified during Inspection of Manhole 2A
CR-PNP-2011-1529, Cables Found Submerged during Manhole Inspections of Manhole 2A, 4, and 5

Section 1R08

PNPS-018-006, Re-inspection of Head to Flange Weld-Magnetic Particle Examination
PNPS-018-023, Linear Indications Noted on Closure Head Washers #19 and 22
PT-11-005, Penetrant Test Summary Sheet for HPCI Vent Valve Installation
MT-RFO18-005, Magnetic Particle Examination Report on Head to Flange Weld
PNPS RFO18-007, Ultrasonic Summary Sheet Safe End to Nozzle 14-A-1
PNPS RFO18-011, Ultrasonic Summary Sheet Safe End to Nozzle 2R-N2G-1
VT-11-174, Visual Examination of IWE Surfaces (Drywell and Torus)
CEP-CII-003 R302, General Visual Examinations of Class MC Components
CEP-NDE-0731 R3, Magnetic Particle Examination (MT) for ASME Section XI

CEP-NDE-0640 R6, Liquid Penetrant Examination (PT) for ASME Section XI
CEP-NDE-0902 R7, VT-2 Examination Report (Visual Examination for Leakage)
1018181 6/09, Nondestructive Evaluations: Guideline for Conducting Ultrasonic Examinations of Dissimilar Metal Welds
2.1.8.7 R7, ASME Code Visual Examination of Primary Containment
TP10-014 R1, Procedure for In Vessel Visual Inspection (IVVI) of BWR 3 RPV Internals
1016645, Nondestructive Evaluation: Procedure for Manual Phased Array Ultrasonic Testing (UT) of Dissimilar Metal Welds (DMW)
INR-P1R1P-IVVI-11-01, RPV Head Closure Studs
INR-P1R1P-IVVI-11-04, Tie Rod Upper Mid Support
INR-P1R1P-IVVI-11-05 R1, Guide Rod Bracket Assembly
INR-P1R18-IVVI-11-06 R1, Steam Separator Lifting Lugs Linear Indications
INR-P1R18-IVVI-11-03, Foreign Material in Vessel
CR-PNP-2011-1796, IVVI Inspection Noted Foreign Material (FME) at Dryer Trough
CR-PNP-2011-1836, IVVI Inspection Noted Foreign Material (Hardware)
CR-PNP-2009-1514, Identified Linear Indication at RPV Head to Flange Weld
CR-PNP-2011-1986, Welded Connection Service Water Support H29-1-2
CR-PNP-2010-2645, ASME Code Classification of Condensate Transfer Piping
CR-PNP-2011-2210, ASME XI IWE General Visual Examination of Various Areas Need Coating Repairs in Drywell and Torus
WR 274898, Repair Broken Weld on Hanger H-29-1-2
WO#5224022519, Perform Weld Repair on RBCCW Heat Exchanger E-209A Channel Wall
WO#52224022512, Perform Visual Examination for Leakage (VT-2)
WO#0024961601, Install Two New Vent Valves in HPCI Suction Piping
PNPS 1.17.4 R1, ASME XI Repair/Replacement Activities
WPS-CU-34/34-C R0, Shielded Metal Arc Welding of Copper Nickel to Itself
WPS-CS-1/1-B R2, Gas Tungsten Arc/Shielded Metal Arc Welding of Carbon Steel
CEP-WP-GWS-1 R1, General Welding Standard ASME/ANSI
PRR-19, Pilgrim Relief Request for weld overlay repair of RPV Jet Pump Instrumentation Nozzle Weld, RPV-N9A-1
M11-81-SH1, Drawing of RBCCW E209A Channel and Wall Detail
EC12422, Drawing of HPCI System 23 of Installed Vent Valves 23-HO-V1 and V2
0995, Certification of Personnel Qualification and Experience Summary Sheets for PDI-UT-10, Rev. C, Addenda 1

Section 1R11

LORT/NRC Simulator Exam Scenario dated 06/01/11 for Loss of TBCCW/HPCI Steam Leak
Licensed Operator Requalification Training Schedule
CR-PNP-2011-3009, NRC Senior Resident identified that drywell CHARMs read 2 R/hr in the plant (downscale lights clear) and 0 R/hr, downscale lights lit in the simulator
PNPS Simulator Discrepancy Report, NRC Resident Inspector observation followed by Control Room walkdown 06/13

Section 1R12

CR-PNP-2009-4468-IRM-D failed to indicate within limits
CR-PNP-2009-4702, IRM-F spiked high several times
CR-PNP-2010-1711, IRM-D meter reading normal with HI and HI-HI lights lit
CR-PNP-2010-2060, IRM-E meter went upscale when drawer was pulled out

CR-PNP-2010-4385, IRM/APRM ‘C’ recorder reading erratically
CR-PNP-2011-0781, Half Scram on IRM-D during start-up
CR-PNP-2011-0782, Half Scram on IRM-B during start-up
CR-PNP-2011-0784, IRM-E recorder select switch does not work
CR-PNP-2011-0794, Received IRM-F HI HI alarm
CR-PNP-2011-0919, NRC question on IRM overlap data acceptance criteria
CR-PNP-2011-1594, IRM/APRM overlap data outside of desired tolerances
CR-PNP-2011-1757, Water in IRM-E shuttle tube
CR-PNP-2011-2468, IRM A/C recorder cannot be selected to Fast Speed
CR-PNP-2011-2477, IRM A Selector Switch difficult to move
CR-PNP-2011-2478, IRM B and IRM H failed to indicate correctly during front panel checks
CR-PNP-2011-2510, Recorder pen not consistently tracking IRM-H
CR-PNP-2011-2564, Half Scram during start-up on IRM-E
CR-PNP-2011-2847, IRM ‘B’ Spiking Causing HI and HI HI alarms
FSAR Chapter 7.5, Neutron Monitoring System
EN-DC-205, Revision 3, Maintenance Rule Monitoring
Neutron Monitoring System Health Report
Neutron Monitoring System Maintenance Rule Basis Document

Section 1R13

Procedure 3.M.1-45, Revision 10, Outage Shutdown Risk Assessment
Procedure 1.5.22, Revision 12, Risk Assessment Process
Procedure 8.M.1-32.6, Revision 32, Analog Trip System Unit Calibration Cabinet C2233A Section B
CR-PNP-2011-1377, Risk profile for Work Week 1114 was potentially non-conservative in regard to 8.M.1-32.6
Refueling Outage (RFO)-18, Shutdown Risk Book
RFO-18, Secondary Containment Contingency
Equipment Out of Service Quantitative Risk Assessment Tool
Work and Test Schedule for Work Week Starting April 4, 2011
Daily Work Plan 6/2/2011
Daily Work Plan 6/6/2011
Schedulers Risk Assessment 6/2/2011

Section 1R15

Procedure 8.M.102, Revision 6, IRM Calibration After controlled Shutdown
Procedure 2.1.5, Revision 111, Controlled Shutdown from Power
Procedure 3.M.3-27, Revision 26, 48V Bus B6 Automatic Transfer Test, UV, Degraded Voltage and Timing Relays Calibration and Annunciator Verification
Procedure 3.M.3-59.1, Revision 5, Transformer Testing
Procedure 8.M.3-1, Revision 55, Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with simulated loss of offsite power and Special Shutdown Transformer Test
Procedure 2.2.23, Revision 33, Automatic Depressurization System
Procedure 8.M.2-3.6.5, Revision 39, Attachment 6, Recirculation Loop Instrumentation Neutron Monitoring Power Range Equipment
Procedure 8.M.1-4, Revision 42, Average Power Monitor Flow Bias Signal Calibration
CR-PNP-2011-1594, IRM/APRM Overlap Data Outside of Procedural Tolerance

CR-PNP-2011-2274, While Performing Bus B6 Transfer System Relay Calibrations, the time delay contacts operated instantaneously
CR-PNP-2011-2024, Post Work Test Results did not meet all acceptance criteria
CR-PNP-2011-2489, Operability Basis for CR-PNP-2011-2188 is not evident
CR-PNP-2011-2556, Error Identified by GE in the Pilgrim Loss of Coolant Accident Analysis Affecting Peak Centerline Temperature
CR-PNP-2011-2188, Acceptance Criteria for 'B' EDG Start and Close on Bus A6 within 10.6 seconds was not met
CR-PNP-2011-2625, C903L-A2 Alarm
CR-PNP-2011-3021, Control Room Vital Area Door Inoperable
CR-PNP-2011-3049, Control Room Vital Area Door Inoperability Not Assessed for Impact on the Control room Envelope
CR-PNP-2011-3007, Intermittent Fault causing Recirculation Flow Converter Failure Alarm
Technical Specifications, Table 4.1.2, Reactor Protection System (SCRAM) Instrument Calibration, Minimum Calibration Frequencies for Reactor Protection Instrument Channels
Technical Specification 3.9, Auxiliary Electrical System
Technical Specification 3/4.S.E, Automatic Depressurization System
Technical Specification 3/4.6.D, Safety and Relief Valves
Technical Specification 3.7.B, Standby Gas Treatment System and Control Room High Efficiency Air Filtration System (CRHEAFS)
FSAR, Section 8.2.2, Preferred AC Power Source Start-up Transformer Test Results
FSAR, Section 8.5, Standby AC Power Source
FSAR, Section 8.4, Auxiliary Power Distribution System
FSAR, Section 4.4, Nuclear System Pressure Relief System
FSAR, Chapter 7.5.7, Average Power Range Monitor Subsystem
General Electric APRM Flow Unit Diagram
Procedure 8.M.2-3.6.5, Revision 37, Attachment 6, Recirculation Loop Instrumentation Neutron Monitoring Power Range Equipment
Procedure 8.M.1-4, Revision 42, Average Power Monitor Flow Bias Signal Calibration
Technical Specification 4.1.2, Reactor Protection System Instrument Calibration
Calculation PS-230, Timing Calculation to Power Emergency Buses during Loss of Coolant
EN-OP-104, Revision 5, Operability Determination Process
EN-OP-104, Revision 5, Operability Evaluation for CR-PNP-2011-2635
ODMI RV-203-3C Leakage

Section 1R18

Procedure 3.M.2-40, Revision 10, Refuel Outage Temporary Modification Reactor Shutdown/ Flood-up Level Indication
Procedure 8.9.1, Revision 118, EDG and Associated Emergency Bus Surveillance: Diesel Generator 'A' Control Room Log
WO#0022012701, Install/Remove Flood-up Level Indication
WO#0022012702, Bench Calibrate Instruments per 3.M.2-40
CR-PNP-2011-2104, Shut Down Level Procedure Revision 10 was issued in March 2011 and work was installed under Revision 9 on April 19, 2011
CR-PNP-2011-2125, Temporary Modification Tag was missing from drawing
CR-PNP-2011-2208, Did not capture lube oil strainer D/P in TP10-13
CR-PNP-2011-1833, Not all As-found data captured
FSAR, Section 4.4, Nuclear System Pressure Relief System
FSAR, Section 8.5, Standby AC Power Source

EN-DC-136, Revision 5, Temporary Modification
EN-DC-115, Revision 11, Engineering Change Process
EN-OP-116, Revision 7, Infrequently Performed Tests or Evolution for PWT of 'A' EDG Governor
EN-WM-105, WO#207002-08, For 'A' EDG Pretest
EN-OP-102-01, Revision 7, Protective and Caution Tagging Forms and Checklist
EN-MA-102, Revision 5, Inspection Program 'A' EDG Governor Modification
EN-DC-134, Revision 3, Design Verification
EN-DC-163, Revision 1, Human Factors Education Form
EN-DC-153, Revision 2, Component Classification Questionnaire
EN-DC-141, Revision 6, Design Input Record
EN-DC-115, Revision 7, Detailed Impact Screening Criteria
EN-LI-100, Revision 9, Process Applicability Determination Form, 50.59 Screening
EC#5974, Engineering Change: EDG Governor
EC 5000071989, Revision 7, SRV/SSV Setpoints and Tolerance Increase and Replacement
Base EC #0000005974, Replace EDG Governor
TP10-013, Revision 0, Special Test for 'A' Governor Replacement Load Performance Post Work Testing

Section 1R19

Procedure 8.9.1, Revision 118, EDG and Associated Emergency Bus Surveillance
Procedure 3.M.3-51, Revision 27, Electrical Termination Procedure
Procedure 8.Q.2-1, Revision 10, Recirculation Water Sample Solenoid Valve (EQ) SV-220-44 Maintenance
Procedure 8.7.1.5, Revision 56, Attachment 59, Local Leak Rate Testing Data Sheet for AO-220-44
Procedure 8.7.4.3, Revision 42, Miscellaneous Containment Isolation Valve Quarterly Operability
Procedure 1.3.34, Revision 119, Attachment 9, Surveillance Test Review for AO-220-44 Closing Time not in Accordance with Procedure 8.7.4.3 Criteria
Procedure 8.I.32, Revision 6, Determination of Limiting Stroke Time Acceptance Criteria for Inservice Testing and Appendix 'B' Test Programs Power-Operated Valves
Procedure 3.M.4-6, Revision 58, Removal, Installation, Test, Disassembly, Inspection and Reassembly of Main Steam Relief Valves
Procedure 8.5.6.2, Revision 37, Special Test for ADS System Manual Opening of Relief Valves
Procedure 8.5.6.4, Revision 14, ADS Operability from Alternate Shutdown Panel
Procedure 3.M.4-1, Revision 36, Control Rod Drive (CRD), Removal and Installation
Procedure 9.9, Revision 66, Control Rod Scram Insertion time Evaluation
Procedure 3.M.3-63.1, Revision 14, Recirculation 'A' MG Set (X-204A) Collector Ring Repair
Procedure 3.M.4-8, Revision 45, Main Steam Isolation Valve Maintenance
Procedure 8.7.1.6, Revision 27, Local Leak Rate Testing of the Main Steam Isolation Valves
WO#5224536401, Replace Reactor Recirculation Sample Valve AO-220-44 Air Supply Solenoid Valve
WO#0022911402, Implement EC 5000071989 for RV-203-3C for RFO-18, Install new Valve
WO#0022914411, PMT-Hydro
WO#0033911412, PMT-Functional
WO#0022911414, PMT-Steam Leak Check
WO#0022911415, Post Maintenance Test – I&C
WO#5224026701, CRD Exchange
WO#5224026702, PMT-CRD Exchange
WO#5224568801, 'A' Recirculation Pump Motor Generator Set (X-204A) Maintenance

WO#5224568802, 'A' Recirculation Pump Motor Generator Set Post Maintenance Test
WO#0023351701, Replace Main Steam Line 'B' Outboard Isolation Valve Actuator
WO#0023351711, PMT-AO-203-2B
CR-PNP-2011-2208, Did not capture lube oil strainer D/P in TP10-13
CR-PNP-2011-1833, Not all as-found data captured
CR-PNP-2011-2151, MSIV 2B Actuator Not Manufactured as Identified on reference Drawing
TP10-013, Revision 0, Special Test for 'A' Governor Replacement Load Performance Post Work Testing
FSAR, Section 8.5, Standby AC Power Source
EN-OP-116, Revision 007, Infrequently Performed Tests or Evolution for PWT of 'A' EDG Governor
EN-WM-105, WO#207002-08, For 'A' EDG Pretest
EN-OP-102-01, Revision 7, Protective and Caution Tagging Forms and Checklist
EN-MA-102, Revision 5, Inspection Program 'A' EDG Governor Modification
EN-DC-134, Revision 3, Design Verification
EN-DC-163, Revision 1, Human Factors Education Form
EN-DC-153, Revision 2, Component Classification Questionnaire
EN-DC-141, Revision 6, Design Input Record
EN-DC-115, Revision 7, Detailed Impact Screening Criteria
EN-LI-100, Revision 9, Process Applicability Determination Form, 50.59 Screening
EC#5974, Engine Change: EDG Governor
Temporary Procedure 11-0000, Revision 0, CRDM Exchange Processes CRD-007 Revision 12 Using: Slim Line Drive Exchange System (SLDES III) or Integrated Drive Exchange Assembly (IDEA) CRDM Exchange Tooling Systems (For use with PNPS 3.M.4-1)

Section 1R20

Procedure 2.1.5, Revision 111, Controlled Shutdown from Power
Procedure 2.1.7, Revision 54, Vessel Heatup and Cooldown
Procedure 3.M.1-45, Revision 11, Outage Shutdown Risk Assessment
Procedure 2.2.19.1, Revision 31, Residual Heat Removal System – Shutdown Cooling Mode of Operation
Procedure 1.5.22, Revision 12, Risk Assessment Process
Procedure 2.1.1, Revision 173, Startup from Shutdown
Procedure 8.A.2, Revision 30, Drywell to Suppression Chamber Vacuum Breaker Leakage Rate Test
EN-OM-123, Revision 3, Fatigue Management Program
EN-FAP-OM-006, Revision 2, Working Hour Limits for Non-Covered Workers
CR-PNP-2011-2434, During torus hatch closeout the nuts for the east hatch could not be located
CR-PNP-2011-2335, Refuel floor observations revealed GE workers not performing a working task
CR-PNP-2011-2442, Control room received unexpected half scram on channel A
CR-PNP-2011-2453, EC5000071989 incorrectly directed the wires to TE-261-40 to be cut
CR-PNP-2011-2438, On 5/8/2011 management in the Outage Control Center instructed the mechanical maintenance supervisor to perform welding for a temporary modification with the documentation to follow
CR-PNP-2011-2419, Planning was directed by the OCC to plan WO#276088 to install temporary mod EC29576 "at risk"
CR-PNP-2011-1661, NRC walkdown noted a rusted fitting (elbow) on the RBCCW cooling water supply piping to MO-4038D/MO-4039D for unit cooler VAC205D

CR-PNP-2011-1667, During vessel disassembly while lowering the reactor head back to the vessel flange, the head suddenly centered itself and slightly shock loaded the hoist
CR-PNP-2011-1670, During reactor head removal noted that in procedure 3.M.4-48.2, Revision 32, Section 8.9.6, Step 3 was incorrect
CR-PNP-2011-1671, During reactor head removal there were several vessel studs that received damage to the upper portion of the stud threads
CR-PNP-2011-2303, Security officer identified that HPCI escape hatch in the 'B' Aux Bay was open and unattached
CR-PNP-2011-2242, There is approximately 6" of standing water contained in the structural steel framing running along the bottom of the N9B bioshield wall
CR-PNP-2011-2206, The GMPO has approved exceeding OT guidelines specified in EN-FAP-OM-006 on Monday May 2 for the Refueling Bridge System Engineer
CR-PNP-2011-1959, Backup IST program engineer was called in by the OCC on their day off assuming GMPO approval. Individual exceeded Guidelines in EN-FAP-OM-006
CR-PNP-2011-1967, Two "opted out" individuals on the Day Shift OCC Exceeded 72 hours in a 7 day period.
CR-PNP-2011-1992, this condition report is in accordance with EN-FAP-OM-006 to document that approval of an opted out of the fatigue rule worker to work his one day off this week with approval per FAP
CR-PNP-2011-2635, SRV-203-3C, 2nd Stage Pilot Valve Leakage
Nuclear Energy Institute (NEI) 06-11, Revision 1, Managing Personnel Fatigue at Nuclear Power Reactor Sites
Qualitative Risk Assessment Notebook
Temporary Procedure (TP) 10-002, Revision 1, RFO18, Compensatory Measures
Technical Specification 3.10, Core Alterations
Technical Specification 3.7.A, Primary Containment
Fatigue Rule CR List
Regulatory Guide 5.73, Fatigue Management for Nuclear Power Plant Personnel
Outage Shutdown Issues
Operations Risk Report
FSAR, Section 5.2, Primary Containment System
Shutdown Schedule
Shutdown Risk Profile

Section 1R22

Procedure 8.9.16.2, Revision 9, Manual Start and Loading of Station Blackout Diesel Generator via Safety bus A5 or A6
Procedure 8.7.3, Revision 59, Secondary Containment Leak Rate Test
Procedure 8.7.1.6, Revision 26, Local Leak Rate Testing of the Main Steam Isolation Valves
Procedure 8.M.3-1, Revision 55, Special Test for Automatic ECLS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loop and Special Shutdown Transformer Load Test
Procedure 8.5.4.6, Revision 40, HPCI Pump and Valve Operability from Alternate Shutdown Panel
Procedure 8.M.1-32.8, Revision 30, Analog Trip System Trip Unit Calibration Cabinet C2233B Section B
Procedure 1.3, 34.7, Revision 18, Data Sheet for RCS Data
Technical Specification 4.9.A, Auxiliary Electrical System
Technical Specification 4.7.C.1, Secondary Containment
Technical Specification 3.6.A.2, Primary System Boundary
Technical Specification 3.7.A.2, Primary Containment Integrity

Technical Specification, Bases 3.5.C, HPCI System
Technical Specification 3.7.A, Containment Systems
Technical Specification 3.12, Alternate Shutdown Panels
Technical Specification 3.5, Core and Containment Cooling System
Technical Specification 3.2.B, Core and Containment Cooling Systems – Limitations and Control
CR-PNP-2011-1519, Data utilized in the secondary containment Leak Rate Test was outside of the acceptance criteria
CR-PNP-2011-2173, During the performance of 8.M.3-1 the LOCA signal did not initiate as expected
CR-PNP-2011-2489, Based on NRC questions regarding the response documentation for CR-PNP-2011-2188 CA 3, the basis for determination for acceptable timing margin for the EDG breaker closing in procedure 8.M.3.1 is not evident from the materials provided or referenced
CR-PNP-2011-2188, During the review of RFO#18, 8.M.301 test result, it was noted that the acceptance criteria [2], page 21 of 77 was not met. The acceptance criteria for EDG 'B' start and close onto A6 bus is within 10.60 seconds.
Drywell Leakage Date Excel Sheet
FSAR, Section 4.6, Main Steam Isolation Valves
FSAR, Section 5.2, Primary Containment System
FSAR, Section 7.3, Primary containment & Reactor Vessel Isolation Control System
FSAR, Section 8.3, Secondary AC Power Source (Shutdown Transformer)
FSAR, Section 8.3.4, Safety Evaluation
FSAR, Section 8.3.5, Inspection and Testing
FSAR, Section 8.5.5, Inspection and Testing
FSAR, Section 8.5.6, Proposed Nuclear Safety Requirement for Initial Plant Operation
FSAR, Section 6.3, Summary Description – Core Standby Cooling Systems
FSAR, Section 6.4, High Pressure Coolant Injection System
FSAR, Section 6.3, Core Standby Cooling Systems
EN-DC-126, Revision 4, Calculation, No. PS230
EN-LI-114, Revision 3, Attachment 9.2, Performance Indicator Process
Sample of Plan of the Day Sheets from Second Quarter 2011 compared to Control Room Log: 3/20/11, 3/17/11, 4/5/11, 5/26/11, 3/14/11, 3/3/11, 2/18/11, 4/11/11, 4/13/11

Section 1EP6

Emergency Planning Performance Indicator Data Submitted
LORT/NRC Simulator Exam Scenario date 06/01/11 for Loss of TBCCW/HPCI Steam Leak

Section 2RS01
EN-RP-101, Revision 5, Access Control for Radiologically Controlled Areas
EN-RP-121, Revision 6, Radioactive Material Control

Section 2RS02
EN-RP-110-01, Revision 0, ALARA Initiative Deferrals
EN-RP-110, Revision 7, ALARA Program
Section 2RS03
EN-RP-131, Revision 8, Air Sampling

Section 4OA1

NRC, Inspection Reports 2nd Quarter 2010 Through 1st Quarter 2011

NRC/NRR PI Website
Procedure EN-LI-114, Revision 4, Performance Indicator Process
CR-PNP-2011-2893

Section 4OA2

Procedure 3.M.4-9, Revision 16, Inspection of the Drywell and Suppression Chamber
Pilgrim Station Quarterly Trend Reports
CR-PNP-2011-0137, Trend of Incomplete | Inaccurate Operability Determinations
CR-PNP-2009-2408, Results of RFO-17 NRC Drywell Inspection
CR-PNP-2009-2414, A Thorough Inspection and Cleanup should have been performed prior to NRC Drywell Inspection
RFO-18, Schedule for Drywell Cleanout and Inspection Activities
CR-PNP-2011-1140, Emerging Trend in Operability Evaluations

Section 4OA3

Procedure 7.4.17, Revision 39, Drywell Continuous Atmospheric Monitoring System
Procedure 7.1.65, Revision 4, Manually Sampling using Panel C41
CR-PNP-2011-3135, C19 A/B Trouble Alarm resulting from CV-5065-92 being closed
Technical Specifications, Section 3.6.C.2, Leakage Detection Systems
Training Schematics on Drywell Leak and Radiation Detection
Drawing P&ID M239, Sh. 1, Revision 28, H2 & O2 Analyzer and Reactor Coolant Pressure Boundary Leak Detection Systems

Section 4OA5

CR-PNP-2010-2645, Corrective Action #18, Clarification required to describe PNPS classification of CST piping

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
ASP	Alternate Shutdown Panel
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
FSAR	Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
IVVI	In-Vessel Visual Inspection
MT	Magnetic Particle Test
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PDI	Performance Demonstration Initiative
PI	Performance Indicator
PNPS	Pilgrim Nuclear Power Station
PT	Liquid Penetrant Test
RBCCW	Reactor Building Closed Cooling Water
RFO	Refueling Outage
RHR	Residual Heat Removal
RPM	Radiation Protection Manager
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWP's	Radiation Work Permit(s)
SLC	Standby Liquid Control
SFP	Spent Fuel Pool
SSC	Structure, System or Component
UT	Ultrasonic Test
VT	Visual Test
WO	Work Order