UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PA 19406-2713

August 11, 2015

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

SUBJECT: PILGRIM NUCLEAR POWER STATION - INTEGRATED INSPECTION REPORT 05000293/2015002

Dear Mr. Dent:

On June 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Pilgrim Nuclear Power Station (PNPS). The enclosed inspection report documents the inspection results, which were discussed on July 22, 2015, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents four NRC-identified findings and one self-revealing finding, four of which are violations of NRC requirements, all of which were of very low safety significance (Green). However, because of the very low safety significance, and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations, consistent with Section 2.3.2.a of the NRC Enforcement Policy. If you contest the non-cited violations in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at PNPS. In addition, if you disagree with the cross-cutting aspect assigned to any finding, or a finding not associated with a regulatory requirement, 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 Resident Inspector at PNPS.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390 of the NRCs "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Raymond R. McKinley, Chief
Reactor Projects Branch 5
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000293/2015002
w/Attachment: Supplementary Information

cc w/encl: Distribution via ListServ

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-293
License No.	DPR-35
Report No.	05000293/2015002
Licensee:	Entergy Nuclear Operations, Inc. (Entergy)
Facility:	Pilgrim Nuclear Power Station
Location:	600 Rocky Hill Road Plymouth, MA 02360
Dates:	April 1, 2015 through June 30, 2015
Inspectors:	E. Miller, Senior Resident Inspector (Acting) E. Carfang, Senior Resident Inspector B. Scrabeck, Resident Inspector E. Knutson, Senior Resident Inspector J. Pfingsten, Reactor Engineer T. Burns, Reactor Inspector B. Dionne, Health Physicist M. Jimenez, Health Physicist J. Furia, Senior Health Physicist S. Pindale, Senior Reactor Inspector
Approved By:	Raymond R. McKinley, Chief Reactor Projects Branch 5 Division of Reactor Projects

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SUMMARY

Inspection Report 05000293/2015002; 04/01/2015 – 06/30/2015; Pilgrim Nuclear Power Station (Pilgrim); Equipment Alignment, Operability Determinations and Functionality Assessments, and Follow-Up of Events and Notices of Enforcement Discretion.

This report covered a three-month period of inspection by resident inspectors and announced inspections performed by regional inspectors. The inspectors identified three non-cited violations (NCVs) and one finding of very low safety significance (Green). The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5.

Cornerstone: Initiating Events

- Green. A self-revealing Green finding was identified when residual heat removal (RHR) pump 'B' experienced cavitation during refueling and maintenance outage (RFO) 20 that was a result of inadequate corrective actions associated with equipment used to determine flow rate. Specifically, prior to placing augmented fuel pool cooling (AFPC) mode in service on April 26, 2015, Entergy did not ensure that the temporary flow transmitter was properly setup and calibrated because corrective actions from 2011 were not adequate to ensure proper setup in the future. As a result, when operators went to raise flow in accordance with their procedural requirement, RHR pump 'B' experienced cavitation and operators secured the pump because the flow transmitter was inaccurately reading low. Entergy's immediate corrective actions included entering the issue into the corrective action program (CAP) as condition report (CR)-2015-3724, re-calibrating and setting up the ultrasonic flow meter, and establishing a second ultrasonic flow meter to ensure proper flow. Inspectors performed a walkdown to ensure proper operation of the ultrasonic flow meters, and confirmed similar readings between the two flow meters on April 27, 2015.

  The finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the 'B' RHR pump was secured from AFPC mode 2 on April 26, 2015 when the installed ultrasonic flow meter did not read properly, leading to operation of the 'B' RHR pump outside of flow limits specified in procedure 2.2.85.2 and cavitation of the pump. This finding was evaluated in accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2, Section C.6 of IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," the inspectors determined that this finding is of very low safety significance (Green) because while the performance deficiency resulted in the 'B' RHR pump being secured due to cavitation, it did occur when the refuel canal/cavity was flooded and did not increase the likelihood of a fire or internal/external flood that could cause an shutdown initiating event. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Evaluation, because Entergy staff did not thoroughly evaluate the issues associated with the ultrasonic flow meter in 2011 and 2013 to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. Specifically, Entergy's corrective action process did not thoroughly evaluate and develop appropriate corrective actions for CR-2011-1847 and CR-2013-2857 to ensure the cause was addressed to prevent challenges using ultrasonic flow meters during AFPC for both mode one and mode two. [P.2] (Section 1R04)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green NCV of Title 10 of the *Code of Federal Regulations* (10 CFR) 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," when Entergy staff performed an inadequate operability determination that assessed the X-107B emergency diesel generator (EDG) following cylinder head leakage indications during pre-start checks for a planned monthly operability run. Specifically, after engine coolant had been observed spraying from one of the open cylinder test cocks during X-107B EDG pre-start checks, operators determined that the EDG remained operable because the volume of leakage that had been observed would not have precluded a successful start of the engine. Operators did not consider that potential sources of leakage, such as a crack in the cylinder or cylinder head, could reasonably worsen during operation, such that the engine would not be able to complete its 30-day mission time, and therefore should be declared inoperable. Entergy's immediate corrective actions included replacement of the X-107B EDG 9L cylinder head and sending out the damaged cylinder head for analysis by a vendor. The completion of the analysis by the vendor is being tracked by CR-2015-2109.

  The finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, Entergy staff inadequately determined that the X-107B EDG was operable, which resulted in the operability of the X-107A EDG not being verified, either through determination that it was not inoperable due to a common cause failure or performing TS SR 4.5.F.1 in its entirety. This finding was evaluated in accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of safety function, did not represent actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding had a cross-cutting aspect in the area of Human Performance, Conservative Bias, because Entergy staff did not use decision making practices that emphasized prudent choices over those that are simply allowed. Specifically, Entergy staff's operability determination for the X-107B EDG was based on the conclusion that the as-found condition would not have caused the engine to be inoperable because it would not have created a hydraulic lock; they did not consider that the condition would likely worsen during EDG operation, nor did their operability determination consider EDG mission time [H.14]. (Section 1R15)

Cornerstone: Occupational/Public Radiation Safety

- Green. The inspectors identified a Green NCV of 10 CFR 20.1406(c) in that Entergy did not conduct operations to minimize the introduction of residual radioactivity on site. Specifically, Entergy did not take action to reduce residual radioactive waste from the site in a timely manner over 14 years for areas in the Radwaste building. Entergy entered this issue into the CAP as CR-2015-5745 with actions to characterize and evaluate the adverse conditions identified by the inspector.

  The finding was more than minor because it is associated with the program and process attribute of the Public Radiation Safety cornerstone and affected the cornerstone objective to ensure the licensee's ability to prevent inadvertent release and/or loss of control of licensed material to an unrestricted area. In accordance with IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process," the finding was determined to be of very low safety significance (Green) because Entergy had an issue involving radioactive material control, but did not involve: (1) transportation; or (2) public exposure in excess of 0.005 Rem. The finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Resolution, in that Entergy did not adequately address the radioactive waste in a 14 year time period. [P.3] (Section 2RS8)

- Green. The inspectors identified a Green NCV of 10 CFR 71.5, "Transportation of Licensed Material," and 49 CFR 172, Subpart I, "Safety and Security Plans." Specifically, Entergy shipped a Category 2 Radioactive Material in Quantities of Concern (RAM-QC) on public highways to a waste processor without adhering to a transportation security plan. Prior to shipment, Entergy's staff failed to recognize that the quantity of radioactive material met the definition RAM-QC. Entergy entered the issue into their CAP as CR-2015-05746 to address changes in Department of Transportation requirements.

  The finding was more than minor because it is associated with the program and process attribute of the Public Radiation Safety cornerstone and affected the cornerstone objective to ensure the safe transport of radioactive material on public highways in accordance with regulations. The finding was determined to be of very low safety significance (Green) because Entergy had an issue involving transportation of radioactive material, but it did not involve: (1) a radiation limit that was exceeded; (2) a breach of package during transport; (3) a certificate of compliance issue; (4) a low level burial ground nonconformance; or (5) a failure to make notifications or provide emergency information. The finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Identification, in that the licensee did not have a low threshold for identifying issues. Specifically, the security transportation plan requirements became effective in March 2003, had not been effectively identified by Entergy. [P.1] (Section 2RS8)

Cornerstone: Miscellaneous

- Severity Level IV. The inspectors identified a Severity Level IV NCV because Entergy personnel did not provide a written report to the NRC within 60 days after discovery of the event as required by 10 CFR 50.73(a)(2)(i)(B) for a condition which was prohibited by TS 3.5.E, "Automatic Depressurization System (ADS)." Specifically, on January 27, 2015, Pilgrim experienced a loss of offsite power and reactor scram during a winter storm. While operators performed a reactor cooldown with manual operation of safety relief valves (SRVs), the 3C SRV twice failed to open upon demand by the operations crew. Entergy staff initiated CR-PNP-2015-0561 to document SRV 3C's failure to open, and the valve was immediately declared inoperable. The inspectors determined that the improper operation of SRV 3C was reportable in accordance with 10 CFR 50.73(a)(2)(i)(B). Entergy has captured this issue in CR-2015-6191.

The inspectors determined that Entergy's failure to submit an event notification in accordance with 10 CFR 50.73 within the required time was a performance deficiency that was reasonably within Entergy's ability to forsee and correct, and should have been prevented. Because the issue had the potential to affect the NRC's ability to perform its regulatory function, the inspectors evaluated this performance deficiency in accordance with the traditional enforcement process. Using example 6.9.d.9 from the Enforcement Policy, the inspectors determined that the violation was a Severity Level IV (a failure of a licensee to make a report required by 10 CFR 50.72 or 10 CFR 50.73) violation. Because this violation involves the traditional enforcement process and does not have an underlying technical violation, inspectors did not assign a cross-cutting aspect to this violation in accordance with IMC 0612, Appendix B. (Section 4OA3)

REPORT DETAILS

Summary of Plant Status

Pilgrim began the inspection period operating at 100 percent reactor power. On April 1, 2015, operators reduced reactor power to 70 percent to perform a rod pattern adjustment and returned to 100 percent the same day. On April 19, 2015, performed a planned reactor shutdown to commence RFO 20. On May 20, 2015, following completion of refueling and maintenance activities, operators took the reactor critical. On May 22, 2015, operators inserted a manual reactor scram due to a degraded condenser vacuum condition. The degraded condenser vacuum condition was the result of a combination of chloride intrusion in the main condenser and water buildup in the augmented offgas system. Following repairs of the system, operators commenced a reactor startup, took the reactor critical and synchronized to the grid on May 23, 2015. On May 27, 2015, operators reduced power to 75 percent to perform a rod pattern adjustment. Operators returned the unit to 100 percent power on the same day. On June 19, 2015, operators reduced reactor power to 30 percent to perform a thermal backwash of the main condenser, circulating water pump bowl cleaning, hydraulic control unit maintenance, and feedwater heater maintenance. Operators returned the unit to 100 percent on June 20, 2015. The unit remained at 100 percent for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 2 samples)

.1 Summer Readiness of Offsite and Alternate Alternating Current (AC) Power Systems

a. Inspection Scope

The inspectors performed a review on June 11, 2015, of Entergy's plant features and procedures for the operation and continued availability of the offsite and alternate AC power system to evaluate readiness of the systems prior to seasonal high grid loading. The inspectors reviewed Entergy's procedures affecting these areas and the communication protocols between the transmission system operator and Entergy. This review focused on changes to the established program and material condition of the offsite and alternate AC power equipment. The inspectors assessed whether Entergy established and implemented appropriate procedures and protocols to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system. The inspectors evaluated the material condition of the associated equipment by interviewing the responsible system manager, reviewing CRs and open work orders (WOs), and walking down portions of the offsite and AC power systems including the switchyard. Documents reviewed for each section of this inspection report are listed in the Attachment.

b. Findings

No findings were identified.

.2 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors performed a review on June 11, 2015, of Entergy's readiness for the onset of the hurricane season. The review focused on the intake structure, the EDGs, the independent spent fuel storage installation, and the station blackout (SBO) diesel generator. The inspectors reviewed station procedures, including Entergy's adverse weather procedures and applicable operating procedures to determine what could challenge these systems, and to ensure Entergy personnel had adequately prepared for these challenges. The inspectors performed walkdowns of the selected systems to ensure station personnel identified issues that could challenge the operability of the systems during adverse weather.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04 – 3 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- 'B' EDG during maintenance on the 'A' EDG and 4.16 kilovolt (kV) bus A5 on April 26, 2015
- 'B' Standby gas treatment (SGT) with 'A' SGT out of service on April 28, 2015
- AFPC on April 27, 2015

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors reviewed applicable operating procedures, system diagrams, the Updated Final Safety Analysis Report (UFSAR), TS, WOs, CRs, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have impacted system performance of their intended safety functions. The inspectors also performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. The inspectors also reviewed whether Entergy staff had properly identified equipment issues and entered them into the CAP for resolution with the appropriate significance characterization.

b. Findings

Introduction. A self-revealing Green finding was identified when RHR pump 'B' experienced cavitation during RFO 20 that was a result of inadequate corrective actions associated with equipment used to determine flow rate. Specifically, prior to placing AFPC mode in service on April 26, 2015, Entergy did not ensure that the temporary flow transmitter was properly setup and calibrated because corrective actions from 2011 were not adequate to ensure proper setup in the future. As a result, when operators went to raise flow in accordance with their procedural requirement, RHR pump ‘B’ experienced cavitation and operators had to secure the pump because the flow transmitter was inaccurately reading low.

Description. Pilgrim’s fuel pool cooling system circulates water through two pumps and two associated heat exchangers in order to maintain fuel pool temperatures acceptably cool during normal operation and refueling. The fuel pool cooling system is designed with an interconnection to the RHR system to either allow for work on certain portions of the reactor system during the RFO or to provide increased cooling capacity.

On April 26, 2015, at 8:35 AM, operators secured normal fuel pool cooling, and at 9:05 AM, operators secured shutdown cooling through ‘B’ RHR system to re-align the system for AFPC mode 2. AFPC mode 2 was used due to the need to perform work on a valve associated with the ‘B’ reactor recirculation loop that required the entire loop to be isolated. Normal shutdown cooling takes suction on the ‘B’ reactor recirculation loop, since this suction path was isolated from the reactor cavity, it was necessary to use AFPC mode 2 to support vessel heat removal during this portion of the outage. At 3:02 PM, operators started the ‘B’ RHR pump for AFPC mode two. At 3:05 PM, operators secured the ‘B’ RHR pump due to the installed ultrasonic flow meter not reading correctly. Procedure 2.2.85.2, “Augmented Fuel Pool Cooling (without shutdown cooling) Mode 2”, Revision 20, directs operators to install the temporary flow meter because the RHR flow indication that operators have in the main control room is not available while in AFPC mode 2. At 4:58 PM, operators started ‘B’ RHR pump again for AFPC; however, operators had challenges controlling level and secured the ‘B’ RHR pump. Following operators assessment for inventory control, operators again started the ‘B’ RHR pump for AFPC mode two.

Procedure 2.2.85.2, Section 5.1[4] states, “The nominal flow rate for AFPC Mode 2 is 1800 gallons per minute (GPM) with one RHR pump in operation. Lower flow rates must be limited to short duration during startup or shutdown to avoid operation below the 1800 GPM long-term minimum flow rate for the RHR pumps. Flow rates above 2000 GPM could cause cavitation on the suction end of the pump.” At 8:29 PM, operators secured the ‘B’ RHR pump due to fluctuating motor amperage, dropping flow and operator field reports of surging sounds. Operators were raising flow to maintain the flow within the procedural limits specified in 2.2.85.2, which led to the pump cavitation. Engineering reviewed the issue with the ultrasonic flow indicator, and determined that the ultrasonic flow transmitter was not properly setup and calibrated to the proper pipe diameter. Following installation of a second ultrasonic flow meter and proper setup of the first ultrasonic flow meter, at 9:26 PM, operators re-established ‘B’ RHR pump for AFPC mode 2, and adjusted flow accordingly to ensure pump flow remained within procedure 2.2.85.2 limits and that the ultrasonic flow meter output was correct.

Entergy generated CR-2015-3782 to document the issues with the ultrasonic flow meter. In addition, CR-2015-3803 was also written to document challenges with initial setup of the ultrasonic flow meter, screening for CR significance, and to note previous challenges with the ultrasonic flow meter as documented in CR-2011-1847 and CR-2013-2857. CR-2015-03803 documented that technicians selected the incorrect pipe diameter during initial setup of the ultrasonic flow meter, leading to erroneous flows during operation. During RFO 18 in 2011, CR-2011-1847 was written to document challenges with the ultrasonic flow meter while placing AFPC mode 1 in service. Operators had noted that flow indications did not match a computer point in the main control room. It was determined that the ultrasonic flow meter was not programmed properly and corrective actions were put in place to update procedures 3.M.2-37, "Temporary Modification Procedure for Fuel Pool Cooling during RFO," and 2.2.85.1, "Augmented Fuel Pool Cooling (Mode 1)." In 2013 during RFO 19, CR-2013-2857 was written to document the ultrasonic flow meter not registering properly when starting AFPC in mode 1. Inspectors noted WO 306297 was written to address the issue, however inspectors identified that the WO was closed due to operators addressing the issue through a procedure change. Inspector interviews identified that although the WO was closed, equipment was not properly connected and therefore did not provide readout to the main control room. Entergy staff immediately updated the procedure, corrected the issue, and connected the equipment. EN-LI-102, "Corrective Action Program," Revision 24, Section 5.5[4](a)(1), states to "ensure actions are assigned as appropriate to correct the Adverse Condition." Contrary to this, following challenges with the ultrasonic flow meter in 2011, issues related to the setup of the meters reoccurred in 2013 and 2015. Inspectors identified that corrective actions from CR-2011-1847 to update AFPC procedures with additional calibration requirements were not incorporated into all AFPC procedures. Specifically, Entergy did not incorporate additional setup and calibration requirements into procedure 2.2.85.2; Entergy only incorporated requirements into 2.2.85.1 and 3.M.2-37.

Entergy's immediate corrective actions included entering the issue into the CAP as CR-2015-3724, re-calibrating and setting up the ultrasonic flow meter, and establishing a second ultrasonic flow meter to ensure proper flow. Inspectors performed a walkdown to ensure proper operation of the ultrasonic flow meters, and confirmed similar readings between the two flow meters on April 27, 2015.

Analysis. The inspectors determined the failure to ensure corrective actions from CR-2011-1847, as required by EN-LI-102, were incorporated in all procedures associated with AFPC to ensure ultrasonic flow meters are properly calibrated prior to starting an RHR pump was a performance deficiency that was within Entergy's ability to foresee and correct and should have been prevented. The finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the 'B' RHR pump was secured from AFPC mode 2 on April 26, 2015 when the installed ultrasonic flow meter did not read properly, leading to operation of the 'B' RHR pump outside of flow limits specified in procedure 2.2.85.2 and cavitation of the pump.

In accordance with IMC 0609.04 Initial Characterization of Findings," and Exhibit 2, Section C.6 of IMC 0609 Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," issued May 9, 2014, the inspectors determined that this finding is of very low safety significance (Green) because while the performance deficiency resulted in the 'B' RHR pump being secured due to cavitation, it did occur when the refuel canal/cavity was flooded and did not increase the likelihood of a fire or internal/external flood that could cause an shutdown initiating event.

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Evaluation, because Entergy staff did not thoroughly evaluate the issues associated with the ultrasonic flow meter in 2011 and 2013 to ensure that resolutions address causes and extent of conditions commensurate with their safety significance. Specifically, Entergy's corrective action process did not thoroughly evaluate and develop appropriate corrective actions for CR-2011-1847 and CR-2013-2857 to ensure the cause was addressed to prevent challenges using ultrasonic flow meters during AFPC for both mode 1 and mode 2.

Enforcement. Enforcement action does not apply because this performance deficiency did not involve a violation of a regulatory requirement. The primary component involved that led the 'B' RHR pump being secured, the ultrasonic flow meter, is not safety related. Entergy has entered the issue into the CAP as CR-2015-3803. Because this finding does not involve a violation, it is identified as a finding **(FIN 05000293/2015002-01, Ineffective Corrective Action leads to Cavitation of Residual Heat Removal Pump).**

.2 Full System Walkdown (71111.04S – 1 sample)

a. Inspection Scope

On May 28 through June 4, 2015, the inspectors performed a complete system walkdown of accessible portions of the Fire Protection Supply System in the intake structure to verify the existing equipment lineup was correct. The inspectors reviewed operating procedures, drawings, equipment line-up check-off lists, system health reports, and the UFSAR to verify the system was aligned to perform its required safety functions. The inspectors also reviewed electrical power availability, component lubrication and equipment cooling, hanger and support functionality, and operability of support systems. The inspectors performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. Additionally, the inspectors reviewed a sample of related CRs to ensure Entergy appropriately evaluated and resolved any deficiencies.

b. Findings

No findings were identified.

1R05 Fire Protection

Resident Inspector Quarterly Walkdowns (71111.05Q – 6 samples)

a. Inspection Scope

The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that Entergy controlled combustible materials and ignition sources in accordance with administrative procedures. The inspectors verified that fire protection and suppression equipment was available for use as specified in the area pre-fire plan, and passive fire barriers were maintained in good material condition. The inspectors also verified that station personnel implemented compensatory measures for out of service, degraded, or inoperable fire protection equipment, as applicable, in accordance with procedures.

- SBO EDG on April 10, 2015
- ‘A’ and ‘B’ Reactor Building Closed Loop Cooling Water (RBCCW) Pump and Heat Exchanger Rooms on April 14, 2015
- Reactor Core Isolation Cooling Pump and Turbine Room on April 14, 2015
- Main Steam and Feedwater Tunnel on April 29, 2015
- Drywell on April 30, 2015
- ‘B’ Train and ‘C’ Train Service Water Pumps Rooms on May 6, 2015

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06 – 1 sample)

Internal Flooding Review

a. Inspection Scope

The inspectors reviewed on April 21–22, 2015, the UFSAR, the site flooding analysis, and plant procedures to assess susceptibilities involving internal flooding. The inspectors reviewed the CAP to determine if Entergy identified and corrected flooding problems and whether operator actions for coping with flooding were adequate. The inspectors also focused on the EDG enclosures to verify the adequacy of equipment seals, penetration seals, watertight door seals, common drain lines, level alarms, and flood barriers as described in the design basis documents.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (711111.07A – 2 samples)

a. Inspection Scope

The inspectors reviewed the E-209A RBCCW heat exchanger and the E-207A RHR heat exchanger to determine their readiness and availability to perform safety functions. The inspectors reviewed the design basis for the components and verified Entergy’s commitments to NRC Generic Letter 89-13. The inspectors performed a visual examination of the heat exchangers and reviewed the results of cleaning and inspections of the heat exchangers. The inspectors reviewed the system health reports and discussed the heat exchanger performance with the responsible system engineer. The inspectors verified that Entergy initiated appropriate corrective actions for identified deficiencies.

- E-209A, RBCCW heat exchanger on April 24, 2015
- E-207A, RHR heat exchanger on April 28, 2015

b. Findings

No findings were identified.

1R08 Inservice Inspection (71111.08G – 1 sample)

a. Inspection Scope

From April 20–28, 2015, the inspectors conducted an inspection of Entergy's implementation of In-Service Inspection (ISI) program activities for monitoring degradation of the reactor coolant system boundary, risk significant piping and components, and containment systems during RFO 20. The sample selection was based on the inspection procedure objectives and risk priority of those pressure retaining components in systems where degradation would result in a significant increase in risk. The inspectors observed portions of in-process non-destructive examinations (NDE), reviewed test procedures, examiner qualification test results, and interviewed Entergy personnel to verify the NDE activities performed were conducted in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2001 edition through 2003 Addenda.

NDE and Welding Activities (IMC Section 02.01)

The inspectors performed a review of current NDE activities in process and associated documents. These reviews included procedure and personnel qualifications to confirm the test examinations were performed in accordance with the requirements of ASME Section XI. Specific documentation inspected was reviewed to verify it had been reviewed and certified by an ASME certified NDE Level III. Activities inspected included review of magnetic particle testing (MT), liquid penetrant testing (PT), ultrasonic testing (UT), and visual examination (VT1 and VT3). In addition, the inspectors observed the remote visual inspection of in-vessel structures and components and evaluated the results for comparison with inspections performed during RFO 18 and 19.

ASME Code Required Examinations

The inspectors observed portions of the manual UT examination of a 10 inch diameter pipe to elbow butt weld in the 'A' RHR system, using examination Procedure GEH-PDI-UT-2, austenitic stainless steel, P8 to P8, WO 378015, Dig ISI-I-10-1, Drywell, component ID 10R-IA-3. Also, the inspectors confirmed the examiner was qualified in accordance with Performance Demonstration Initiative EPRI-DMW-PA-1 and ASME Section XI, Appendix VIII. No reportable indications were identified.

The inspectors performed a documentation review of the automated UT examination of reactor recirculation piping loop 'A', austenitic piping weld (reducer to pipe) 2R-N1B-1 (reference drawing ISI I 2R-A R6). The welding variables were identified as 28 inch diameter dissimilar metal weld of nozzle to safe end. The examination was conducted in accordance with Automatic UT procedure CEP-NDE-0496. The inspectors reviewed the automatic welding process and non-destructive ultrasonic test by observation of the welding of a "mock up" representing the actual component configuration, materials, and welding technique to be used. No reportable indications were identified.

The inspectors performed a documentation review of the PT examination of four support lugs installed on the 'B' core spray pipe support (reference drawing H14-1-33, Component GB-14-22HL1(4)). The examination was accomplished using Procedure CEP-NDE-0641, on WO 377771.

No reportable indications were identified.

The inspectors performed a documentation review of the fabrication of replacement pipe spool JF29-9-5, low carbon steel, rubber lined and installed in the service water system by welding under WO 00362953-03. The inspectors determined the replacement spool was fabricated, inspected, and installed to drawing M100-7251 requirements of ASME (B&PV) Section XI. Two welds were selected for examination using the MT examination procedure CEP-NDE-0731. No reportable indications were identified.

Other Augmented, License Renewal, or Industry Initiative Examinations

The inspectors performed a review of the results of the current remote visual examination of selected in-vessel components which included structural members, piping, supports, and restraints within the reactor pressure vessel. The in-vessel video records of previous indications, their location, characterization, and evaluation were made during this RFO. The inspection scope included upper portions of the core shroud, steam dryer, core spray piping inside the vessel, and portions of other structural members. The inspection determined that Entergy performed these activities in accordance with applicable industry guidance.

Containment Visual Examination

The inspectors visually examined the condition of the primary containment liner at the 23 foot elevation and limited portions of locations above and below that elevation for the entire 360 degrees of the inside diameter. The inspectors utilized portable lighting to view the liner surfaces to assess the surface condition. The inspectors noted that the condition of the liner plate and the liner coating reflected evidence that the liner was being maintained in serviceable condition with some areas having been locally re-coated to provide the base metal with protection from degradation. The inspectors reviewed photographic documentation of restored areas and performed a review of the previous (RFO 19) IWE VT-1 and VT-3 examinations for purposes of comparison to the current liner condition.

Review of Originally Rejectable Indications Accepted by Evaluation

There was no ASME B&PV Code, Section XI NDE indications from previous outages that required follow-up inspection during RFO 20.

Repair/Replacement Consisting of Welding Activities

For component replacement work, the inspectors reviewed the fabrication, inspection, and replacement instructions for salt service water system spool piece JF29-9-5. Fabrication by welding was in accordance with drawing M100-7251 and WO 00362953-03. The work instruction package, including the requirements for weld procedure specifications, weld filler metals, and non-destructive testing were reviewed. Applicable quality certifications and verifications were reviewed for compliance with requirements of ASME B&PV, Section XI.

Identification and Resolution of Problems (IMC Section 02.05)

The inspectors reviewed a sample of Pilgrim CRs, which identified NDE indications and other non-conforming conditions since the previous RFO and during the current outage. The inspectors verified that non-conforming conditions were properly identified, characterized, evaluated, and corrective actions were identified and entered into the CAP for resolution.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11 – 2 samples)

.1 Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on June 8, 2015, which included two scenarios. The first involved an electric grid warning, ‘A’ reactor recirculation motor generator set overspeed, fuel failure, a reactor water cleanup leak, and a radiation release in the turbine building and reactor building. The second scenario involved a loss of the Y-1 power supply, a loss of the A6 4.16kV safety-related electrical bus, and an SBO. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors verified the accuracy and timeliness of the emergency classification made by the shift manager and the TS action statements entered by the shift technical advisor. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

.2 Quarterly Review of Licensed Operator Performance in the Main Control Room

a. Inspection Scope

For the plant activities listed below, the inspectors observed and reviewed operator performance in the main control room. See section 4OA3 for specific discussion of these activities. The inspectors reviewed operational and alarm response and implementation of procedural guidance. The inspectors also observed control room conduct and control of evolutions and events, in accordance with procedure EN-OP-115, “Conduct of Operations,” Revision 14.

- Rod pattern adjustment on April 1, 2015
- Reactor shutdown on April 19, 2015
- Reactor startup activities on May 20, 2015 – May 23, 2015

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q – 2 samples)

a. Inspection Scope

The inspectors reviewed the samples listed below to assess the effectiveness of maintenance activities on structure, system, or component (SSC) performance and reliability. The inspectors reviewed system health reports, CAP documents, maintenance WOs, and maintenance rule (MR) basis documents to ensure that Entergy was identifying and properly evaluating performance problems within the scope of the MR. For each sample selected, the inspectors verified that the SSC was properly scoped into the MR in accordance with 10 CFR 50.65 and verified that the (a)(2) performance criteria established by Entergy staff was reasonable. As applicable, for SSCs classified as (a)(1), the inspectors assessed the adequacy of goals and corrective actions to return these SSCs to (a)(2). Additionally, the inspectors ensured that Entergy staff were identifying and addressing common cause failures that occurred within and across MR system boundaries.

- Drywell floor and equipment sump monitoring systems on April 6-14, 2015
- Fire Protection Supply System on June 1-9, 2015

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 6 samples)

a. Inspection Scope

The inspectors reviewed station evaluation and management of plant risk for the maintenance and emergent work activities listed below to verify that Entergy performed the appropriate risk assessments prior to removing equipment for work. The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that Entergy personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When Entergy performed emergent work, the inspectors verified that operations personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work to verify plant conditions were consistent with the risk assessment. The inspectors also reviewed the TS requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

- Planned elevated risk during testing of the analog trip system on April 3, 2015
- Planned elevated risk due to the low pressure coolant injection (LPCI) system being inoperable for valve testing on April 17, 2015
- Planned elevated risk due to maintenance on the 4.16kV A5 electrical bus on April 24, 2015

- Planned elevated risk with the ‘A’ SGT treatment system inoperable on April 27, 2015
- Loss of the shutdown transformer (SDT) and SBO EDG on May 2, 2015
- Planned elevated risk with 345kV Line 355 out of service to support grid maintenance on June 9, 2015

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 – 6 samples)

a. Inspection Scope

The inspectors reviewed operability determinations for the following degraded or non-conforming conditions:

- Failure of the drywell floor sump low level alarm and resultant excessive sump pumpdown on April 4, 2015
- Salt service water pump ‘E’ discharge piping on April 10, 2015
- Degraded bus bar insulation for unit auxiliary transformer and start-up transformer electrical feed to 4.16kV A5 electrical bus on April 14, 2015
- X-107B EDG Cylinder Head 9L on April 28, 2015
- 10-HO-1001-33A ‘A’ reactor recirculation manual suction isolation valve packing leakage on May 5, 2015
- Drywell to Suppression Chamber Vacuum Breakers on June 5, 2015

The inspectors selected these issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the operability determinations to assess whether TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria from the TSs and UFSAR to Entergy’s evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled by Entergy. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” when Entergy staff performed an inadequate operability determination that assessed the X-107B EDG following cylinder head leakage indications during pre-start checks for a planned monthly operability run. Specifically, after engine coolant had been observed spraying from one of the open cylinder test cocks during X-107B EDG pre-start checks, operators determined that the EDG remained operable because the volume of leakage that had been observed would not have precluded a successful start of the engine. Operators did not consider that potential sources of leakage, such as a crack in the cylinder or cylinder head, could reasonably worsen during operation, such that the engine would not be able to complete its 30-day mission time, and therefore should be declared inoperable.

<u>Description</u>. On March 18, 2015, at 2:15 AM, operators entered TS 3.5.F, “Minimum Low Pressure Cooling and Diesel Generator Availability,” to perform pre-startup checks of the X-107B EDG in accordance with procedure 8.9.1, “Emergency Diesel Generator and Associated Emergency Bus Surveillance,” Revision 129. TS 3.5.F provides a 72 hour limiting condition for operation (LCO) that can be extended to 14 days provided that all low pressure core and containment cooling systems, and the SBO diesel generator are determined to be operable. When the engine was rolled over with air to verify that no fluid was present in any of the cylinders, engine coolant was instead observed to spray out of the open cylinder test cock on cylinder 9L. Entergy staff estimated that approximately six ounces of fluid was discharged. This issue was entered into the CAP as CR-2015-02109. Entergy staff determined that the X-107B EDG had been and remained operable because the volume of fluid that had been discharged would not have produced a hydraulic lock on cylinder 9L and therefore would not have prevented the engine from starting. Entergy staff exited TS 3.5.F at 2:30 AM.

On March 18, 2015, at 9:16 AM, Entergy staff determined that an inspection of cylinder 9L should be performed, and entered TS 3.5.F. Initial troubleshooting was inconclusive as to where the leak was coming from, leading Entergy staff to exit TS 3.5.F and prepare additional troubleshooting plans. At 4:00 PM, Entergy staff entered TS 3.5.F to continue troubleshooting and perform additional inspections of the cylinder head. The scope of this activity subsequently expanded to include replacement of the associated cylinder head. In discussions with the inspectors, Entergy staff stated that the condition did not render the EDG inoperable, but that they were entering voluntary LCOs for the purpose of investigation and troubleshooting only. Entergy staff performed surveillance procedure 8.9.16.1, “Manually Start and Load Blackout Diesel via the Shutdown Transformer,” Revision 48, at 5:40 PM, to extend the TS 3.5.F allowed outage time to 14 days. Testing of the replaced head showed the source of the leakage to have been from the area of the cylinder exhaust valves. Entergy’s immediate corrective actions included replacement of the X-107B EDG 9L cylinder head and sending out the damaged cylinder head for analysis by a vendor. The completion of the analysis by the vendor is being tracked by CR-2015-2109. Entergy staff exited TS 3.5.F following successful post-maintenance testing at 6:11 AM on March 21, 2015. From identification of the issue through correction of the problem by replacement of the 9L cylinder head, Pilgrim staff maintained that the condition had not caused the X-107B EDG to be inoperable. Entergy staff stated that their EDGs were capable of operating with one cylinder removed from service; however, were unable to provide the inspectors with any design documents or engineering calculations showing that the EDGs would be capable of supplying design basis loads under such conditions.

The inspectors reviewed CR-2015-02109 and the associated apparent cause evaluation (ACE). While the inspectors agreed that the as-found condition would not have prevented the X-107B EDG from starting, they did not conclude that the EDG remained operable. Although the source of the engine coolant leak was unknown at the time of discovery, it could reasonably have been due to a crack in the cylinder head. Such a leak would have the possibility of worsening during engine operation. Although hydraulic locking of the cylinder would not be a realistic concern during engine operation, increased engine coolant leakage into the cylinder would result in water intrusion into the crankcase and lubricating oil sump, which would eventually cause the engine to fail.

Therefore, the inspectors concluded that the X-107B EDG should have been declared inoperable after engine coolant had been identified in cylinder 9L.

Entergy procedure EN-OP-104, "Operability Determination Process," Revision 9, states that, for an immediate operability determination, "if a piece of information material to the determination is missing or unconfirmed, and cannot reasonably be expected to support a determination that the SSC [structure, system, or component] is OPERABLE, the SM (shift manager) should declare the SSC INOPERABLE." In this case, at the time of discovery, although the cause of the leak had not been established, it could reasonably have been due to a crack in the cylinder head. For the reasons discussed above, it could be concluded that this condition would not support a determination that the X-107B EDG remained operable. Additionally, an operability determination example presented in Attachment 9.1, "Operability Classification Guide," of this procedure indicates that an EDG that cannot run for the duration assumed in the current licensing basis should be considered inoperable. SDBD-61, "Design Basis Document for Emergency Diesel Generator (EDG)," states, "The 'mission time' for the design basis Loss-of-Coolant-Accident (LOCA) is 30 days for the long term containment cooling analysis, as described in TDBD100 "Design Basis Document for Design Basis Accidents, Transients and Special Events (DBATS)." Therefore, the inspectors further concluded that Pilgrim staff also should reasonably have concluded that the X-107B EDG should have been declared inoperable after engine coolant had been identified in cylinder 9L.

TS 3.5.F, "Minimum Low Pressure Cooling and Diesel Generator Availability," provides a 72 hour allowed outage time for one EDG, provided the remaining EDG is demonstrated to be operable per TS SR 4.5.F.1. TS SR 4.5.F.1 requires that, within 24 hours, a determination be made that the operable EDG is not inoperable due to a common cause failure, or that the monthly TS-required surveillance test be performed for the operable EDG, and that, within 1 hour and every 8 hours thereafter, correct breaker alignment and indicated power availability for each offsite circuit be verified. If these requirements cannot be met, TS 3.5.F further requires that the reactor be placed in cold shutdown within 24 hours. Since Entergy staff did not declare the X-107B EDG inoperable as a result of the engine coolant leakage issue, but instead entered what Entergy staff considered to be voluntary LCOs for the purpose of investigation, only the portion of TS SR 4.5.F.1 for offsite breaker verification was performed. Therefore, the inspectors additionally concluded that Entergy staff's failure to perform the required determination that the operable EDG was not inoperable due to common cause failure constituted a violation of TS 3.5.F.

The TS-required monthly surveillance test was satisfactorily completed on the X-107A EDG on April 2, 2015, approximately two weeks after the X-107B EDG 9L cylinder head coolant leakage event. While this did not eliminate the TS violation discussed above, it did demonstrate that, from a risk perspective, the X-107A EDG had been capable of performing its design safety function during that period.

<u>Analysis</u>. The inspectors determined that Entergy's inadequate operability determination of the X-107B EDG after engine coolant was found in one of the cylinders, and resultant failure to determine that the X-107A EDG was not inoperable due to a common cause failure, or to perform the complete TS-specified EDG monthly surveillance test, within 24 hours in accordance with TS SR 4.5.F.1, was a performance deficiency that was within Entergy's ability to foresee and correct, and should have been prevented. The finding was more than minor because it was associated with the equipment performance

attribute of the Mitigating Systems cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, Entergy staff inadequately determined that the X-107B EDG was operable, which resulted in the operability of the X-107A EDG not being verified, either through determination that it was not inoperable due to a common cause failure or performing TS SR 4.5.F.1 in its entirety.

In accordance with IMC 0609.04, "Initial Characterization of Findings," and Exhibit 2 of IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," the inspectors determined that this finding was of very low safety significance (Green) because the performance deficiency was not a design or qualification deficiency, did not involve an actual loss of safety function, did not represent actual loss of a safety function of a single train for greater than its TS allowed outage time, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event.

This finding had a cross-cutting aspect in the area of Human Performance, Conservative Bias, because Entergy staff did not use decision making practices that emphasized prudent choices over those that are simply allowed. Specifically, Entergy staff's operability determination for the X-107B EDG was based on the conclusion that the as-found condition would not have caused the engine to be inoperable because it would not have created a hydraulic lock; they did not consider that the condition would likely worsen during EDG operation, nor did their operability determination consider EDG mission time [H.14].

Enforcement. 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that "activities affecting quality shall be prescribed by documented instructions, procedures, or drawings… and shall be accomplished in accordance with these instructions, procedures, or drawings." Procedure EN-OP-104, "Operability Determination Process," Revision 9, states, in part, that "if a piece of information material to the determination is missing or unconfirmed, and cannot reasonably be expected to support a determination that the SSC [structure, system, or component] is OPERABLE, the SM (shift manager) should declare the SSC INOPERABLE." Also, during any period when one EDG is inoperable, TS 3.5.F allows continued reactor operation during the succeeding 72 hours, provided that the remaining EDG is demonstrated to be operable in accordance with TS SR 4.5.F.1. TS SR 4.5.F.1 requires that, within 24 hours, a determination be made that the operable EDG is not inoperable due to a common cause failure, or that the monthly surveillance test be performed on the operable EDG in accordance with TS SR 4.9.A.1.a, and that, within 1 hour and once every 8 hours thereafter, correct breaker alignment and indicated power availability for each offsite circuit be verified. If this requirement cannot be met, then the reactor shall be placed in the cold shutdown condition within 24 hours.

Contrary to the above, on March 18, 2015, Entergy staff performed an inadequate operability determination of the X-107B EDG following indications of engine coolant leakage in cylinder 9L, the X-107A EDG was not demonstrated to be operable in accordance with TS SR 4.5.F.1, in that a determination that the X-107A EDG was not inoperable due to a common cause failure was not made, nor was the monthly surveillance test performed on the X-107A EDG in accordance with TS SR 4.9.A.1.a. Because this violation was of very low safety significance (Green) and Entergy staff entered this issue into their CAP as CR-2015-2109, this violation is being treated as a NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000293/2015002-02, Inadequate Operability Determination for the X-107B EDG Results in TS Violation)**

1R19 Post-Maintenance Testing (71111.19 – 6 samples)

a. Inspection Scope

The inspectors reviewed the post-maintenance tests for the maintenance activities listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the test procedure to verify that the procedure adequately tested the safety functions that may have been affected by the maintenance activity, that the acceptance criteria in the procedure was consistent with the information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed test data to verify that the test results adequately demonstrated restoration of the affected safety functions.

- Replacement of X-107A EDG starting air and turbo air assist air check valves on March 3–5, 2015
- Repair of MO-1001-34A RHR Loop A Torus Cooling/Spray Block Valve on May 1, 2015
- Overhaul of the P-208B Salt Service Water pump on May 4, 2015
- Control rod drive pump P209A and P209B following troubleshooting on May 8, 2015
- Overhaul of the AO-203-1D Main Steam Isolation Valve on May 16, 2015
- G23 vital motor generator set on June 11, 2015

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20 – 1 sample)

a. Inspection Scope

The inspectors reviewed the station's work schedule and outage risk plan for the maintenance and refueling outage (RFO 20) which commenced on April 19, 2015. The inspectors reviewed Entergy's development and implementation of outage plans and schedules to verify that risk, industry experience, previous site-specific problems, and defense-in-depth were considered. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored controls associated with the following outage activities:

- Configuration management, including maintenance of defense-in-depth, commensurate with the outage plan for the key safety functions and compliance with the applicable TSs when taking equipment out of service
- Implementation of clearance activities and confirmation that tags were properly hung and that equipment was appropriately configured to safely support the associated work or testing

- Reactor cooldown and heatup activities
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and instrument error accounting
- Status and configuration of electrical systems and switchyard activities to ensure that TSs were met
- Monitoring of decay heat removal operations
- Reactor water inventory controls, including flow paths, configurations, alternative means for inventory additions, and controls to prevent inventory loss
- Refueling activities
- Reactor cavity draindown
- Fatigue management
- Tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block the emergency core cooling system suction strainers, and startup and ascension to full power
- Identification and resolution of problems related to refueling outage activities

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 7 samples)

a. Inspection Scope

The inspectors observed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether test results satisfied TSs, the UFSAR, and Entergy procedure requirements. The inspectors verified that test acceptance criteria were clear, tests demonstrated operational readiness and were consistent with design documentation, test instrumentation had current calibrations and the range and accuracy for the application, tests were performed as written, and applicable test prerequisites were satisfied. Upon test completion, the inspectors considered whether the test results supported that equipment was capable of performing the required safety functions. The inspectors reviewed the following surveillance tests:

- Secondary containment leak rate test on April 15, 2015
- Manual start and loading of the SBO diesel generator of safety bus A5 on April 23, 2015
- As-Found local leak rate testing of the Main Steam Isolation Valves on April 24-25, 2015 (CIV)
- Automatic emergency core cooling system load sequencing of diesels and SDT with simulated loss of offsite power and special SDT load test on May 11, 2015
- Scram discharge volume vent and drain isolation valve timing on May 15, 2015
- Special test for ADS system manual opening of relief valves on May 21, 2015 (IST)
- High pressure cooling injection (HPCI) system pump and valve quarterly and biennial comprehensive operability test on May 21, 2015 (IST)

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational/Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01 - 1 sample)

a. Inspection Scope

During May 4 - May 7, 2015, the inspectors reviewed Entergy's performance in assessing and controlling radiological hazards in the workplace during RFO 20. The inspectors used the requirements contained in 10 CFR 20, TSs, Regulatory Guide (RG) 8.38, and the procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed applicable radiation protection procedures, radiation protection program audits and reports of operational occurrences in occupational radiation safety since the last inspection.

Radiological Hazard Assessment

The inspectors reviewed recent plant radiation surveys and any changes to plant operations since the last inspection to identify any new radiological hazards for onsite workers or members of the public.

Instructions to Workers

The inspectors observed several containers of radioactive materials and assessed whether the containers were labeled and controlled in accordance with requirements.

The inspectors reviewed several occurrences where a worker's electronic personal dosimeter alarmed. The inspectors reviewed Entergy's evaluation of the incidents, documentation in the CAP, and whether compensatory dose evaluations were conducted, when appropriate.

Contamination and Radioactive Material Control

The inspectors observed the monitoring of potentially contaminated material leaving the radiological control area and inspected the methods and radiation monitoring instrumentation used for control, survey, and release of that material.

Radiological Hazards Control and Work Coverage

The inspectors evaluated in-plant radiological conditions and performed independent radiation measurements during facility walk-downs and observation of radiological work activities. The inspectors assessed whether posted surveys, radiation work permits (RWPs), worker radiological briefings, the use of continuous air monitoring, and dosimetry monitoring were consistent with the current radiological conditions. The inspectors examined the control of highly activated or contaminated materials stored within the spent fuel pools. The inspectors evaluated posting and physical controls for selected high radiation areas (HRAs), locked high radiation areas (LHRAs) and very high radiation areas (VHRA) to verify conformance with the applicable regulations and licensing agreements.

<u>Risk-Significant HRA, LHRA, and VHRA Controls</u>

The inspectors reviewed the controls and procedures for HRAs, LHRAs, VHRAs, and radiological transient areas in the plant.

<u>Problem Identification and Resolution</u>

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were identified at an appropriate threshold and properly addressed in the CAP.

b. <u>Findings</u>

No findings were identified.

2RS2 <u>Occupational ALARA Planning and Controls</u>

a. <u>Inspection Scope</u>

The inspectors assessed Entergy's performance with respect to maintaining occupational individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors used the requirements contained in 10 CFR 20, RG 8.8 and RG 8.10, TSs, and procedures required by TSs as criteria for determining compliance.

<u>Inspection Planning</u>

The inspectors conducted a review of Pilgrim's collective dose history and trends, ongoing and planned radiological work activities, radiological source term history and trends, and ALARA dose estimating and tracking procedures.

<u>Radiological Work Planning</u>

The inspectors selected for review the following radiological work activities based on risk significance:

- PNPS-RWP-2015-481, Scaffolding RFO 20
- PNPS-RWP-2015-542, Torus Desludge RFO 20
- PNPS-RWP-2015-0509, Exchange CRDs and Support Activities
- PNPS-RWP-2015-530, Work Inside A & B Condensers – Dog Bone Replacement

For each of these activities, the inspectors reviewed: ALARA work activity evaluations; exposure estimates; exposure/contamination reduction techniques; dose estimates and results achieved; and person-hour estimates and results achieved; as well as in-progress ALARA reviews that were conducted on these work activities.

Verification of Dose Estimates and Exposure Tracking Systems

The inspectors reviewed the current annual collective dose estimate, basis methodology, and measures to track, trend, and reduce occupational doses for ongoing work activities.

Source Term Reduction and Control

The inspectors reviewed the current plant radiological source term and historical trend, plans for plant source term reduction, and contingency plans for changes in the source term as the result of changes in plant fuel performance or changes in reactor water or condensate water chemistry.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with ALARA planning and controls were identified at an appropriate threshold and properly addressed in the CAP.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation

a. Inspection Scope

The inspectors reviewed the control of in-plant airborne radioactivity and the use of respiratory protection devices in these areas. The inspectors used the requirements in 10 CFR 20, RG 8.15, RG 8.25, NUREG-0041, TS, and procedures required by TS as criteria for determining compliance.

Inspection Planning

The inspectors reviewed the UFSAR to identify ventilation and radiation monitoring systems associated with airborne radioactivity controls and respiratory protection equipment. The inspectors also reviewed respiratory protection program procedures and current performance indicators for unintended internal exposure incidents.

Engineering Controls

The inspectors reviewed operability and use of both permanent and temporary ventilation systems, and the adequacy of airborne radioactivity radiation monitoring in the plant based on location, sensitivity, and alarm set-points.

Use of Respiratory Protection Devices

The inspectors reviewed the adequacy of Entergy's use of respiratory protection devices in the plant to include applicable ALARA evaluations, respiratory protection device certification, respiratory equipment storage, air quality testing records, and individual respirator qualification records.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with the control and mitigation of in-plant airborne radioactivity were identified at an appropriate threshold and addressed by Entergy's CAP.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment

a. Inspection Scope

The inspectors reviewed the monitoring, assessment, and reporting of occupational dose. The inspectors used the requirements in 10 CFR 20; RGs 8.7, 8.9, 8.26, 8.34, 8.38 and 8.40; TSs; and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors reviewed: radiation protection program audits, National Voluntary Laboratory Accreditation Program (NVLAP) dosimetry testing reports, and procedures associated with dosimetry operations.

External Dosimetry

The inspectors reviewed: dosimetry NVLAP accreditation, onsite storage of dosimeters, the use of "correction factors" to align electronic personal dosimeter results with NVLAP dosimetry results, dosimetry occurrence reports, and CAP documents for adverse trends related to external dosimetry.

Internal Dosimetry

The inspectors reviewed: internal dosimetry procedures, whole body counter measurement sensitivity and use, adequacy of the program for whole body count monitoring of plant radionuclides, adequacy of the program for dose assessments based on air sample monitoring and the use of respiratory protection, and internal dose assessments for any actual internal exposure greater than 10 millirem.

Special Dosimetric Situations

The inspectors reviewed: Entergy's worker notification of the risks of radiation exposure to the embryo/fetus, the dosimetry monitoring program for declared pregnant workers, external dose monitoring of workers in large dose rate gradient environments, and dose assessments performed since the last inspection that used multi-badging or skin dose.

Problem Identification and Resolution

The inspectors evaluated whether problems associated with occupational dose assessment were identified at an appropriate threshold and properly addressed in the CAP.

b. Findings

No findings were identified.

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation (71124.08 – 1 sample)

a. Inspection Scope

The inspectors verified the effectiveness of Entergy's programs for processing, handling, storage, and transportation of radioactive material. The inspectors used the requirements of 49 CFR 170-177; 10 CFR 20, 37, 61, and 71; applicable industry standards; RGs; and procedures required by TSs as criteria for determining compliance.

Inspection Planning

The inspectors conducted an in-office review of the solid radioactive waste system description in the UFSAR, the process control program, and the recent radiological effluent release report for information on the types, amounts, and processing of radioactive waste disposed. The inspectors reviewed the scope of quality assurance audits performed for this area since the last inspection.

Radioactive Material Storage

The inspectors observed radioactive waste container storage areas and verified that Entergy had established a process for monitoring the impact of long-term storage of the waste.

Radioactive Waste System Walk-down

The inspectors walked down the following items and areas:

- Accessible portions of liquid and solid radioactive waste processing systems to verify current system alignment and material condition
- Abandoned in place radioactive waste processing equipment to review the controls in place to ensure protection of personnel
- Changes made to the radioactive waste processing systems since the last inspection
- Processes for transferring radioactive waste resin and/or sludge discharges into shipping/disposal containers
- Current methods and procedures for dewatering waste

Waste Characterization and Classification

The inspectors identified radioactive waste streams and reviewed radiochemical sample analysis results to support radioactive waste characterization. The inspector's reviewed the use of scaling factors and calculations to account for difficult-to-measure radionuclides.

Shipment Preparation

The inspectors reviewed the records of shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness.

Shipping Records

The inspectors reviewed selected non-excepted package shipment records.

Identification and Resolution of Problems

The inspectors assessed whether problems associated with radioactive waste processing, handling, storage, and transportation, were identified at an appropriate threshold and properly addressed in Entergy's CAP.

b. Findings

1. Introduction. The inspectors identified a Green NCV of 10 CFR 20.1406(c) in that Entergy did not conduct operations to minimize the introduction of residual radioactivity on site. Specifically, Entergy did not take action to reduce residual radioactive waste from the site in a timely manner over 14 years for areas in the Radwaste building.

Description. Entergy has multiple areas holding residual radioactive waste over a 14 year period. The conditions resulted in the dispersal of highly radioactive material in the rooms for at least two years. The rooms are controlled as LHRAs, limiting exposure to plant personnel and waste was confined to the rooms. The following locations contain highly radioactive material that has not been addressed in a timely manner:

- Radwaste building (elevation -13') – resin on spent resin tank room floor
- Radwaste building (elevation -3') – resin and sludge on the clean waste tanks room floor
- Radwaste building (elevation -3') – abandoned radioactive waste concentrator
- Reactor building (elevation 51') – backwash receiver tank room floor drain backflow

Based on NRC inspection of the areas in 2000 and 2013, no effective corrective actions to reduce waste levels in these areas occurred. CR-2013-1221 was written to address the inspector's concerns in 2013; however, the actions taken to address the concern were ineffective. The drains were hydrolazed clean and resin removed; however, radioactive waste was still present during the NRC inspection in June 2015. Entergy entered this issue into their CAP as CR-2015-5745 with actions to characterize and evaluate the adverse conditions identified by the inspector.

Analysis. The failure to minimize residual radioactive waste is a performance deficiency within Entergy's ability to foresee and correct, and should have been prevented. Specifically, radioactive waste was not minimized due to Entergy's failure to address long standing issues of radioactive waste from overfill of tanks, drains, and the abandonment of the concentrator resulting in the dispersal of highly radioactive material in those rooms over a 14 year period. The issue is more than minor because it is associated with the program and process attribute of the Public Radiation Safety cornerstone and affected the cornerstone objective to ensure the licensee's ability to prevent inadvertent release and/or loss of control of licensed material to an unrestricted area. In accordance with IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process," the finding was determined to be of very low safety significance (Green) because Entergy had an issue involving radioactive material control, but did not involve: (1) transportation; or (2) public exposure in excess of 0.005 Rem.

The finding has a cross-cutting aspect in the area of Problem Identification and Resolution, Resolution, in that effective corrective actions were not taken to address issues in a timely manner commensurate with their safety significance. Specifically, Entergy did not adequately address the radioactive waste in a timely manner. The conditions existed over a 14 year period and had not been adequately addressed [P.3].

Enforcement. 10 CFR 20.1406(c) requires, in part, that licensees shall, to the extent practical, conduct operations to minimize the introduction of residual radioactivity on site. Contrary to the above, from 2000 to the present, Entergy did not, to the extent practical, conduct operations to minimize the introduction of residual radioactivity on site. Specifically, Entergy failed to adequately maintain the radwaste systems and address residual radioactive waste, causing the introduction of residual radioactivity to the site. Because this violation is of very low safety significance (Green) and Entergy entered this issue into their CAP as CR-2015-5745, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(05000293/2015002-03, Failure to Conduct Operations to Minimize the Introduction of Residual Radioactivity to the Site)**.

2. Introduction. The inspectors identified a Green NCV of 10 CFR 71.5, "Transportation of Licensed Material," and 49 CFR 172, Subpart I, "Safety and Security Plans." Specifically, Entergy shipped a Category 2 RAM-QC on public highways to a waste processor without adhering to a transportation security plan. Prior to shipment, Entergy's staff failed to recognize that the quantity of radioactive material met the definition RAM-QC.

   Description. Entergy prepared a radioactive waste liner for shipment to a radioactive waste processor. The liner, containing spent resin, was determined to have a total activity of 625 curies including 120 curies of cobalt (Co-60) as indicated on the Uniform Low-Level Radioactive Waste Manifest (NRC Form 541) which Entergy had generated. The liner was shipped on August 30, 2012. The inspectors reviewed the shipment and determined that the shipment met the definition of Category 2 RAM-QC since it contained more than 8.1 curies of Co-60. The liner was shipped without Entergy implementing the required transportation security plan.

   Entergy initiated CR-2015-05746, with corrective actions that included revising the shipping procedure to reflect the appropriate Department of Transportation requirements for shipment of Category 2 RAM-QC.

Analysis. The failure to ship material as a Category 2 RAM-QC was a performance deficiency that was reasonably within Entergy's ability to foresee and correct, and should have been prevented. The issue is more than minor because it is associated with the program and process attribute of the Public Radiation Safety cornerstone and affected the cornerstone objective to ensure the safe transport of radioactive material on public highways in accordance with regulations. In accordance with IMC 0609, Appendix D, "Public Radiation Safety Significance Determination Process," the finding was determined to be of very low safety significance (Green) because Entergy had an issue involving transportation of radioactive material, but it did not involve: (1) a radiation limit that was exceeded; (2) a breach of package during transport; (3) a certificate of compliance issue; (4) a low level burial ground nonconformance; or (5) a failure to make notifications or provide emergency information.

The finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Identification, in that Entergy did not have a low threshold for identifying issues. Specifically, the security transportation plan requirements became effective in March 2003, had not been effectively identified by Entergy [P.1].

Enforcement. 10 CFR 71.5, "Transportation of Licensed Material," requires compliance with the applicable requirements of Department of Transportation regulations in 49 CFR Parts 171 through 180. 49 CFR 172, Subpart I, "Safety and Security Plans," [49 CFR 172.800(b)] requires that known radionuclides in forms listed as Category 2 RAM-QC must adhere to a transportation security plan. Contrary to the above, on August 20, 2012, Entergy did not comply with requirements that Category 2 RAM-QC must adhere to a transportation security plan. Specifically, Entergy made a Category 2 shipment of RAM-QC (Shipment ID 12-09) without providing or adhering to a transportation security plan. Because this violation is of very low safety significance (Green) and Entergy entered this issue into their CAP as CR-2015-05746, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(05000293/2015002-04, Failure to Properly Ship Category 2 Radioactive Material – Quantity of Concern)

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 – 3 samples)

a. Inspection Scope

The inspectors reviewed performance indicator data to determine the accuracy and completeness of the reported data. The review was accomplished by comparing reported performance indicator data to confirmatory plant records and data available in plant logs, CRs, licensee event reports (LERs), and NRC Inspection Reports. The acceptance criteria used for the review was Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guidelines," Revision 7. The following performance indicators were reviewed:

- Unplanned Scrams per 7000 Critical Hours
- Unplanned Scrams with Complications
- Unplanned Power Changes per 7000 Critical Hours

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 – 3 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

As required by Inspection Procedure 71152, “Problem Identification and Resolution,” the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that Entergy entered issues into their CAP at an appropriate threshold, gave adequate attention to timely corrective actions, and identified and addressed adverse trends. In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the CAP and periodically attended CR review group meetings.

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of site issues, as required by Inspection Procedure 71152, “Problem Identification and Resolution,” to identify trends that might indicate the existence of more significant safety issues. In this review, the inspectors included repetitive or closely-related issues that may have been documented by Entergy outside of the CAP, such as trend reports, performance indicators, major equipment problem lists, system health reports, MR assessments, and maintenance or CAP backlogs. The inspectors also reviewed Entergy’s CAP database for the first and second quarters of 2015 to assess CRs written in various subject areas (equipment problems, human performance issues, etc.), as well as individual issues identified during the NRCs daily CR review (Section 4OA2.1). The inspectors reviewed the Entergy quarterly trend reports to ensure that Entergy’s personnel were appropriately evaluating and trending adverse conditions in accordance with applicable procedures.

b. Findings and Observations

No findings were identified.

Corrective Action Program

As a follow up to the previous semi-annual trend review the inspectors reviewed CR-2014-2740 in which Entergy conducted an ACE to address adverse trends associated with the CAP. The inspectors assessed effectiveness and sustainability of corrective actions. The ACE evaluated various weaknesses identified from July 2013 to July 2014 that included the following:

- CR-2013-4577 – weaknesses with the performance of nine ACEs
- CR-2013-4995 – actions from previous problem identification and resolution assessments were not effective at resolving corrective action closure quality and timeliness of corrective actions
- CR-2013-5391 – quality assurance audit of the CAP program was identified as marginally effective due to continued issues with corrective action closeout timeliness and quality, CAP backlog with open CRs, and Corrective Action Review Board backlog
- CR-2013-7830 – NRC performance indicators for reactor scrams crossing the green-to-white threshold and the relationship to the CAP by not effectively being used to ensure effective corrective actions were taken to prevent subsequent reactor scrams
- CR-2014-1669 – common cause evaluation of weaknesses in nuclear safety culture associated with CAP
- CR-2014-3762 – elevation and escalation of Nuclear Oversight Concerns due to weaknesses in corrective action closures and timeliness continuing at the station despite actions taken over the past year to correct them

The ACE targeted specific areas of the CAP which included CR initiation, CR screening and assignment, CR evaluation and resolution, and CR closure. An evaluation by Entergy determined the largest weaknesses in executing the CAP were associated with performing the evaluation and resolution of a CR along with the closure process. A gap analysis associated with "organizational and programmatic evaluation" determined the lack of a formal accountability structure at Pilgrim which influenced weaknesses in culpability for enforcing the responsibilities with quality and challenging the execution of corrective actions. The direct cause of the ACE was determined to be a lack of effective leadership engagement which resulted in low quality execution of the CAP, with respect to evaluation, resolution, and closure. The apparent cause was determined to be a failure to apply a structured accountability platform to ensure the needed changes in behavior had developed for improvements in execution of the CAP.

In the time since CR-2014-2740, inspectors noted continued challenges associated with effectiveness of the CAP as documented in NRC Inspection Report 05000293/2014008 and captured by Entergy in CR-2015-0375. Given that corrective actions from CR-2014-2740 were closed out and showed weaknesses in sustainability, Entergy staff confirmed that CR-2015-0375 would continue to implement many of the corrective actions from CR-2014-2740 and continue to improve the gaps identified in the CAP. The inspector's review noted that at the time of the inspection the corrective action review board backlog had been reduced to zero since 2013. In September 2014, EN-LI-102, "Corrective Action Program," Revision 24, was released. The release revised how Entergy classified conditions identified in the plant by now dividing them between being an adverse condition and non-adverse condition. EN-LI-102 defines an "Adverse Condition" as a general term which includes Conditions Adverse to Quality plus condition related to the following: (a) design basis, (b) licensing basis, (c) NRC regulations and commitments, (d) State and Federal regulations other than NRC, (e) key elements of reactor oversight process, and (f) equipment required to support safety-related equipment as defined by the functionality assessment process in EN-OP-104. Although Entergy has shown a reduction in the CR backlog, this new process changed how Entergy was tracking CR backlogs. Based on the new process, Entergy is now tracking with a reduced backlog of CRs because the process focuses on adverse CRs. Thus, although recent numbers show a reduction in CR backlog, it is primarily due to a change in tracking process.

CR-2015-0375 performed an ACE that evaluated causes for failing the NRC 95002 Supplemental Inspection. The ACE determined two causes for not meeting the objectives of the NRC 95002 supplemental inspection. The first was that Pilgrim leadership had not fully aligned the organization to internalize that CAP is a core business element essential for the continued improvement of Pilgrim, through communication, modeling of behaviors, and use of the accountability model. It was determined to be due to an unbalanced reward system (positive and negative) that places insufficient value on successful implementation of the CAP. The second cause determined by Entergy was that existing weaknesses in the organization's ability to oversee and monitor performance created unforeseen flaws in the 95002 project plan resulting in inadequate preparation for the 95002 pre-inspection. Entergy implemented corrective actions that included the use of departmental performance improvement coordinators for review of closed corrective actions for quality gaps, providing the departmental performance improvement coordinators specific training and mentoring on execution of their responsibilities in CR screening, classification, action closure, and trending. It also included a number of other corrective actions which included additional apparent and root cause training, evaluation of Culpability Reviews with the Site Vice President, the General Manager of Plant Operations, and the Engineering Director to identify departments that are not demonstrating CAP excellence in their category. Performance goals were also established as part of the 2015 Performance Review process with a focus on organizational ownership and effective use of the CAP. Inspectors reviewed the ACE and corrective actions in more detail during the 95002 follow-up inspection and results were documented in Inspection Report 05000293/2014008 (ML15026A069). The corrective actions appear to be reasonable and inspectors will continue to follow progress through daily review of the CAP and periodic trend review to determine if corrective actions are effective and sustainable.

<u>Annunciators</u>

The inspectors noted a trend associated with reduced reliability of the Beta annunciator system. The Beta annunciator system provides alarm indication in the main control room to alert reactor operators of off-normal conditions with plant systems. Between January 2015 and June 2015 there have been 12 CRs that have been issued regarding deficiencies and challenges associated with the Beta annunciator system. Also within the same time frame, three separate losses of the Beta annunciator system occurred. The first on February 16, 2015, as documented in CR-2015-1201; the second on May 1, 2015, as documented in CR-2015-4106; and the third on May 5, 2015, as documented in CR-2015-4393. Each of these losses occurred when the reactor was in a cold shutdown condition and did not impact Emergency Action Levels for event response. However, the trend displays the challenges associated the Beta annunciator system and with the CAP program and its ability to address deficiencies that date back to July 15, 2013, when a complete loss of the Beta annunciator system occurred that resulted in a declaration of an Unusual Event and a 50.72(a)(1)(i) event report (#49189) to the NRC, as documented in CR-2013-5208.

Inspectors identified during the 95002 inspection in November 2014 that the last replacement of the backup power card was not captured in the root cause evaluation.

Entergy documented this in CR-2014-6096. In March 2015, CR-2015-1761 documented that CR-2013-5208 did not identify a definitive root cause and that vendor analysis results were inconclusive regarding a potential failure of a power supply. CR-2015-1761 also noted that the extent of cause and extent of condition was narrowly focused. To resolve CR-2015-1761, corrective action CA-42 associated with CR-2013-5208 was designated to address the deficiencies. With regard to extent of condition and extent of cause, CA-42 performed a review of various systems however; focus was again limited to only systems that could cause entry into an emergency operating procedure or emergency action level. Interviews with Entergy staff identified that CR-2013-5208 will be reviewed as part of corrective actions associated with CR-2015-0375 to identify weaknesses associated with extent of condition and extent of cause. Entergy documented this in CR-2015-4788. Additional corrective actions since the previous failures include adjustment of the voltage to the power supplies in accordance with vendor recommendations and planned implementation of preventative maintenance actions as specified by the recent failure associated with annunciator terminal system in CR-2015-4106. The inspectors will continue to follow this trend and assess effectiveness of corrective actions to address issues associated with the Beta annunciator system.

.3 Annual Sample: Through-wall Leak of Feedwater Heater Shell E-103B

a. Inspection Scope

The inspectors performed a review of Entergy's root cause evaluation and corrective actions associated with CR-2014-4052. The inspectors assessed the problem identification threshold, extent of condition reviews, and the prioritization and timeliness of corrective actions to determine whether Entergy personnel were appropriately identifying, characterizing, and correcting problems associated with the history of the erosion/corrosion of numerous locations of the E-103B feedwater heater shell.

Specifically, feedwater heater E-103B has a history of shell erosion/corrosion issues. Since 1999, these issues have been resolved by a series of repairs and evaluations performed as needed to extend the replacement date of the heater. The Flow Accelerated Corrosion Program has measured and monitored the thickness of portions of the shell since 1999. No tubes have been plugged (removed from service) since heater installation in 1984. Feedwater heater E-103B was removed and replaced during the current refuel outage (RFO 20) with an identical heater procured without modification or change to original specifications. The replacement feedwater heater was procured for a planned replacement due to the continued degradation and resultant leaks in the shell.

The inspectors assessed Entergy's problem identification threshold, cause analyses, extent of condition reviews, compensatory actions, and the prioritization and timeliness of Entergy's corrective actions to determine whether Entergy staff were appropriately, identifying, characterizing, and correcting problems associated with this issue and, whether the planned and/or completed corrective actions were appropriate. The inspectors took note that the feedwater heater while being risk significant is not a safety-related component. However, the feedwater heater was designed and fabricated to the requirements of ASME Section VIII. In addition, the inspectors interviewed responsible engineering personnel to assess the effectiveness of the implemented corrective actions.

b. Findings and Observations

No findings were identified.

The direct cause of the shell leaks was erosion of shell base metal and penetration in areas around the shell structural stiffeners until those areas could no longer carry the required loads. The inspectors reviewed test records of ultrasonic thickness readings acquired to support that wall thickness met the acceptance levels contained in engineering documents. However, wall thinning and loss of adjacent structural support to the shell and stiffeners continued.

The inspectors reviewed Entergy’s causal evaluation that identified the likely causes of the failure, including the degradation by corrosion of the shell in numerous area locations and in the areas of shell structural support. This condition resulted in the development of a susceptible condition resulting in a failure by shell leakage of the heater during operation on August 15, 2014. The inspectors noted that a replacement for the E-103B heater had been purchased and placed into storage in 2000.

The inspectors confirmed the critical parameters were being tracked and included appropriate alert and action levels when wall thinning decreased to a “t-critical” level. This wall thickness provides design margin with an inclusive allowance such that action is taken prior to actual encroachment on the design minimum allowable wall thickness.

The inspectors also reviewed a selection of sample locations where the highest wear rates have been detected in heater wall thickness monitoring plans. The inspectors interviewed engineering staff and reviewed test data to verify that monitoring and trending of wear data was measured and evaluated by engineering personnel.

The inspectors reviewed a selection of test data for various components and did not identify any additional issues. The inspectors determined Entergy’s overall response to this issue was commensurate with the safety significance, was timely, and included reasonable compensatory actions. The inspectors concluded that actions completed were reasonable to correct the problem and prevent reoccurrence.

.4 Annual Sample: Safety Relief Valve Temporary Modification

a. Inspection Scope

The inspectors performed an in-depth review of Entergy’s actions associated with replacing the previously installed 3-stage SRVs with 2-stage SRVs. Specifically, the plant response to the loss of offsite power and reactor scram event on January 27, 2015, was complicated by several equipment performance issues, including the failure of the ‘C’ SRV to open upon manual actuation. Following disassembly, the valve’s manufacturer, Curtiss-Wright Flow Control Company, Target Rock Division, issued a 10 CFR Part 21 Interim Report on March 17, 2015, due to the potential to induce a defect during the testing of the relief valve model (3-stage Target Rock Model 0867F). In particular, the loads induced on the test stand can cause the main disk to piston and main disk to locknut preload to be lost on the 3-stage SRVs. Loss of preload allows vibration induced fretting between the piston rings and liner; and over time, this fretting can increase piston friction to the point where valve functionality is challenged. On May 1, 2015, the manufacturer provided an update to that report, which stated that the root cause was not yet determined. As additional challenges were apparent with the design and testing of the 3-stage SRVs, Entergy elected to remove all four 3-stage SRVs and replace them with 2-stage SRVs as an interim measure while they and the manufacturer continued to evaluate and correct issues with the 3-stage SRVs.

The inspectors reviewed documents and interviewed engineering personnel to assess the acceptability and effectiveness of their actions and adequacy of the associated temporary modification. The inspectors also reviewed the associated CRs, including those that were related to issues identified with the replacement 2-stage SRVs. In addition, the inspectors independently reviewed Entergy's responses to manufacturer correspondence to evaluate the adequacy of their actions.

b. Findings and Observations

No findings were identified.

The inspectors determined that Entergy's overall response to the 3-stage SRV design and/or testing deficiencies was commensurate with the safety significance. Since the cause(s) of the deficient SRV performance have not been positively identified, Entergy elected to replace the 3-stage SRVs with 2-stage SRVs of a design in use at other industry boiling water reactors and previously installed at Pilgrim. Available 2-stage SRVs were refurbished and supplied to the station to support the implementation of the temporary modification during RFO 20. Entergy plans to use the 2-stage SRVs as an interim measure until the 3-stage SRV issues are resolved.

The inspectors found that the temporary modification and associated design of the 2-stage SRVs were consistent with the design and licensing bases. The inspectors concluded that SRV critical parameters and characteristics remained acceptable and did not represent new or unanalyzed challenges. For comparison purposes, the inspectors also reviewed the prior modification that originally installed the 3-stage SRVs and an SRV lift setpoint TS change request and associated NRC Safety Evaluation Report. Inspectors also observed surveillance testing of the 2-stage SRV's during startup from RFO 20, and observed proper operation when actuated manually from the main control room. The inspectors concluded that the 2-stage SRV design did not invalidate any existing commitments or requirements.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 3 samples)

.1 Plant Events

a. Inspection Scope

For the plant events listed below, the inspectors reviewed and/or observed plant parameters, reviewed personnel performance, and evaluated performance of mitigating systems. The inspectors communicated the plant events to appropriate regional personnel, and compared the event details with criteria contained in IMC 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of potential reactive inspection activities. As applicable, the inspectors verified that Entergy made appropriate emergency classification assessments and properly reported the event in accordance with 10 CFR Parts 50.72 and 50.73. The inspectors reviewed Entergy's follow-up actions related to the events to assure that Entergy implemented appropriate corrective actions commensurate with their safety significance.

- Manual reactor scram due to degraded condenser vacuum on May 22, 2015

b. Findings

No findings were identified.

.2 (Closed) LER 05000293/2015-001-00: Loss of 345 KV Power Resulting in Automatic Reactor Scram during Winter Storm Juno

a. Inspection Scope

The inspectors reviewed Entergy's actions and reportability criteria associated with LER 05000293/2015-001-00, which is addressed in CR-2015-0558. On January 27, 2015, Pilgrim experienced a loss of offsite power and reactor scram during a winter storm. The cause of the event was determined to be that the design of the Pilgrim switchyard does not prevent flashover when impacted by certain weather conditions experienced during severe winter storms. The event was complicated by the failure of the K-117 diesel powered air compressor, which resulted in the loss of the instrument air system. Additionally, while operators performed a reactor cooldown with manual operation of SRVs, the 3C SRV twice failed to open upon demand by the operations crew. Associated findings are contained in NRC Special Inspection Report 05000293/2015007. Prior to restoration of offsite power to the switchyard, the switchyard bus insulators and bushings were cleaned of snow and salt contamination to prevent further flashovers. Station procedures have been revised to provide additional guidance including the requirement to place the reactor in cold shutdown prior to severe winter storms. Planned corrective actions are to implement a switchyard design change to minimize switchyard flashovers during snowstorms. The inspectors review identified a Severity Level IV traditional enforcement NCV for the failure to provide a LER within 60 days. The enforcement actions associated with this LER are discussed below. This LER is closed.

b. Findings

Introduction. The inspectors identified a Severity Level IV NCV because Entergy personnel did not provide a written report to the NRC within 60 days after discovery of the event as required by 10 CFR 50.73(a)(2)(i)(B) for a condition which was prohibited by TS 3.5.E, “Automatic Depressurization System (ADS).” Specifically, on January 27, 2015, when operators attempted to depressurize the reactor vessel using SRV 3C, the relief valve did not open to lower reactor pressure as required.

Description. The Pilgrim reactor vessel pressure relief system includes two safety valves and four relief valves. The safety valves provide protection from overpressure of the reactor vessel and discharge directly to the interior space of the drywell. The relief valves, which discharge to the suppression pool, provide three main functions; overpressure relief operation to limit pressure rise and prevent spring safety valve opening; overpressure safety operation to prevent reactor vessel overpressurization; and depressurization operation in automatic or manual by control room operators as part of the ADS. The purpose of the ADS system is to serve as a backup to the HPCI system under loss of coolant accident conditions. If the HPCI system does not operate and one of the LPCI or core spray pumps is available, the reactor vessel is depressurized sufficiently to permit the LPCI and core spray systems to inject water into the vessel to protect the fuel barrier. TS 3.5.E states, in part, that ADS shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 104 psig and prior to a startup from a cold condition, with an allowed outage time of 14 days after the date that one valve in the ADS system is made or found to be inoperable for any reason, provided that during those 14 days the HPCI system is operable. If those requirements cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within twenty four hours.

At 4:02 AM on January 27, 2015, Pilgrim experienced a loss of offsite power and reactor scram during a winter storm. While operators performed a reactor cooldown with manual operation of SRVs, the 3C SRV twice failed to open upon demand by the operations crew. At 10:15 AM, an open signal as applied to SRV 3C for 52 seconds, reactor pressure increased from 220 psig to 224 psig, and there was no corresponding change in reactor water level. At 10:32 AM, a second attempt was made, and an open signal was applied for 83 seconds, reactor pressure increased from 262 psig to 266 psig, and again there was no corresponding change in reactor water level. After the second failure of SRV 3C, plant operators continued with the plant cooldown using SRVs 3B and 3D. Due to pre-existing second stage leakage and concerns about the ability to reclose the SRV after use, SRV 3A was not utilized. Additionally, the HPCI system had already been declared inoperable at 9:53 AM the same day due to the loss of instrument air drain lines and the resultant overflowing of water through its gland seal condenser blower. Entergy staff initiated CR- 2015-0561 to document SRV 3C's failure to open, and the valve was immediately declared inoperable. While the reactor was shutdown, SRV 3C was removed from the plant and transported to a third party vendor for testing. The vendor successfully performed inservice testing at the required high pressure, and subsequently performed a successful test at low pressure. Based on the test results, on February 5, 2015, Entergy staff revised their determination of operability and declared that the SRV 3C had remained operable for the entirety of the event.

Based on the valve failing to operate on demand when the manual switch was twice taken to the open position, the absence of supporting plant indications of expected response to valve operation, and indications of fretting on the main piston cylinder due to loosening of the stem nut discovered during the initial disassembly and inspection, the inspectors determined that the vendor testing was not sufficient to provide reasonable assurance that while installed SRV 3C remained operable at low pressures, and therefore the inspectors challenged Entergy's revised operability determination. Following the inspectors challenge, Entergy declared SRV 3C inoperable on March 16, 2015. Findings associated with Entergy's operability determinations for the SRV 3C are contained in NRC Special Inspection Report 05000293/2015007. Although the final determination of inoperability was not made until March 16, the indications to determine the failure of SRV 3C were readily available on January 27, 2015, and were recognized by the control room staff, resulting in the immediate declaration of inoperability. The inspectors determined that the improper operation of SRV 3C was reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition which was prohibited by the plant's TS, and the guidance contained in NUREG-1022, "Event Report Guidelines 10 CFR 50.72 and 50.73," which states, in part, that the discovery date is the date when the event was discovered rather than the date when the condition is reviewed, and that if additional evaluation is undertaken that a report should be made when reasonable expectation of operability no longer exists or significant doubts begin to arise. LER 2015-001-00, issued on April 1, 2015, documented the loss of offsite power and made various required reports related to the event; however, the LER did not include the required report for the failure of SRV 3C. The failure of SRV 3C was ultimately reported in LER 2015-002-00, issued on May 12, 2015. Entergy has captured this in CR-2015-6191.

Analysis. The inspectors determined that Entergy's failure to submit a written report within 60 days in accordance with 10 CFR 50.73(a)(2)(i)(B) was a performance deficiency that was within Entergy's ability to foresee and correct and should have been prevented. Because the issue had the potential to affect the NRC's ability to perform its regulatory function, the inspectors evaluated this performance deficiency in accordance with the traditional enforcement process. Using example 6.9.d.9 from the Enforcement Policy, the inspectors determined that the violation was a Severity Level IV (a failure of a licensee to make a report required by 10 CFR 50.72 or 10 CFR 50.73) violation. Because this violation involves the traditional enforcement process and does not have an underlying technical violation, inspectors did not assign a cross-cutting aspect to this violation in accordance with IMC 0612, Appendix B.

Enforcement. 10 CFR 50.73(a)(2)(i)(B) requires, in part, that licensees shall report any operation or condition which is prohibited by the plant's TS within 60 days of discovering the event. Contrary to the above, Entergy did not submit a report within 60 days of January 27, 2015, after it was discovered that the SRV 3C was not operable as required by TS 3.5.E. Specifically, this condition was not reported until May 12, 2015, 105 days after the event, under LER 2015-002-00. Because this issue was of Severity Level IV and has been entered into Entergy's CAP under CR-2015-6191, this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. **(NCV 05000293/2015002-05, Failure to Submit an LER).**

.3 (Closed) LER 05000293/2015-002-00: Main Steam Safety Relief Valves Determined to be Inoperable Following Evaluation

The inspectors reviewed Entergy's actions and reportability criteria associated with LER 05000293/2015-002-00, which is addressed in CR-2015-0561, CR-2015-1520, and CR-2015-1983. On January 27, 2015, while responding to a loss of offsite power and reactor scram, SRV 3C failed to open upon demand by plant operators for plant cooldown. Further inspection revealed internal damage to the main stage piston section of the SRV. Additionally, the extent of cause review revealed a similar failure of SRV 3A to operate during a plant cooldown in 2013. Findings associated with Entergy's failure to identify the 2013 failure of SRV 3A are contained in NRC Special Inspection Report 05000293/2015007.

Entergy determined during testing in February 2015 that fretting wear between the main stage piston and liner lead to increased friction in the stroke of the valve. Additional investigation is still underway by the manufacturer of the SRVs, and any new causal information will be communicated to Entergy and other applicable licensees as determined by the 10 CFR Part 21 reporting process. As an immediate corrective action, Entergy replaced SRV 3A and SRV 3C, prior to restart from the forced outage in January 2015. During RFO 20, all SRVs were removed and replaced with 2-stage SRVs of a design that has not shown similar degradation. The replacement of the 2-stage SRVs was reviewed separately as part of a problem identification and resolution sample and is documented in section 4OA2 of this report. Additional corrective actions are captured in CR-2015-0561. The inspectors determined that the report of the failure of SRV 3C was not timely in accordance with the requirement of 10 CFR 50.72. The enforcement actions associated with this LER are discussed with the closure of LER-2015-001-00 in section 4OA3 of this report. This LER is closed.

4OA6 Meetings, Including Exit

On July 22, 2015, the inspectors presented the quarterly baseline inspection results to Mr. John Dent, Site Vice President, and other members of the Pilgrim staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Dent	Site Vice President
G. Blankenbiller	Chemistry Manager
T. Bordelon	Performance & Improvement Manager
P. Beabout	Security Manager
K. Bienvenue	Code & Programs Engineer
S. Brewer	Radiation Protection Supervisor
G. Blankenbiller	Chemistry Manager
R. Byrne	Senior Licensing Engineer
D. Calabrese	Emergency Preparedness Manager
B. Chenard	Engineering Director
F. Clifford	Operations Support Manager
S. Asplin	Senior System and Components Engineer
S. Brewer	Radiation Protection Supervisor
J. Cotter	Operations Training Supervisor
P. Doody	Senior Design Engineer
P. Harizi	Senior Design Engineer
M. Jacobs	Manager of Nuclear Oversight
M. Landry	Senior Systems and Components Engineer
C. Littleton	Senior Lead Design Engineer
J. Macdonald	Senior Operations Manager
E. McCaffrey	System and Components Engineering Supervisor
R. McGaha	Code & Programs NDE Services
R. Morris	Senior System and Components Engineer
J. Moylan	Manager, Project & Maintenance Services
D. Noyes	Director of Regulatory & Performance Improvement
J. O'Donnell	Senior System and Components Engineer
J. Ohrenberger	Senior Maintenance Manager
E. Perkins	Regulatory Assurance Manager
R. Pardee	Code & Programs Engineer
R. Passalugo	Shipper
M. Perry	Systems Engineer
N. Reece	System and Components Engineer
M. Rose	NDE Services-Level III
J. Sabina	IST Program Engineer
M. Thornhill	Radiation Protection Supervisor
D. Tkatch	Radiation Protection Manager
S. Verrochi	General Manager Plant Operations
A. Zelie	Radiation Protection Manager

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

05000293/2015002-01	FIN	Ineffective Corrective Actions leads to Loss of Decay Heat Removal (Section1R04)
05000293/2015002-02	NCV	Inadequate Operability Determination for the ‘B’ EDG Results in TS Violation (Section 1R15)
05000293/2015002-03	NCV	Failure to Conduct Operations to Minimize the Introduction of Residual Radioactivity to the Site (Section 2RS8)
05000293/2015002-04	NCV	Failure to Properly Ship Category 2 Radioactive Material – Quantity of Concern (Section 2RS8)
05000293/2015002-05	NCV	Failure to Submit an LER (Section 4OA3)

Closed

05000293/2015-001-00	LER	Loss of 345 KV Power Resulting in Automatic Reactor Scram During Winter Storm Juno (Section 4OA3)
05000293/2015-002-00	LER	Main Steam Safety Relief Valves Determined to be Inoperable Following Evaluation (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
1.4.4, New England Power Grid Operations/Interfaces, Revision 27
1.5.22, Risk Assessment Process, Revision 25
2.1.14, Station Power Changes, Revision 113
2.4.144, Degraded Voltage, Revision 42
5.3.31, Station Blackout, Revision 18
5.9.1, Extended Loss of AC Power (ELAP), Revision 0
2.1.37, Coastal Storm – Preparations and Actions, Revision 38
8.C.40, Seasonal Weather Surveillance, Revision 31
2.1.42, Operation during Severe Weather, Revision 26
ENN-PL-158, Transmission Grid Interface and Compliance with NERC Standards, Revision 9

Condition Reports

CR-2012-0907	CR-2013-0798	CR-2015-0558

Work Order

52378103	52459584	52371646	00180907
52378102	52374132		

Miscellaneous
Final Safety Analysis Report (FSAR), Section 8.5, Standby AC Power Source
FSAR, Section 8.10, Blackout AC Power Source
NRC GL-2006-02, Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power
Regulatory Guide 1.155 Station Blackout

Section 1R04: Equipment Alignment

Procedures
2.1.12.1, Emergency Diesel Generator Surveillance, Revision 81
2.2.50, Standby Gas Treatment, Revision 68
2.2.19.1, Residual Heat Removal System – Shutdown Cooling Mode of Operation, Revision 39
2.2.85.2, Augmented Fuel Pool Cooling (Without Shutdown Cooling) Mode 2, Revision 20
2.2.25, Fire Water Supply System, Revision 59

Drawings
M294, Heating Ventilation & Air Conditioning Standby Gas Treatment System Control Diagram, Revision 27
M231, Fuel Pool Cooling and Demineralizer System, Revision 44
M241 Sh1, Residual Heat Removal System, Revision 87
M241 Sh2, Residual Heat Removal System, Revision 87
M218 Sh2, P&ID Fire Protection System, Revision 46
E9, Single Line Meter & Relay Diagram 480V System – Load Centers & motor Control CentersB10 & B20, Revision 67

Condition Reports (*NRC Identified)

CR-2015-0621	CR-2015-0727	CR-2015-0791
CR-2015-0795	CR-2015-0935	CR-2015-0936
CR-2015-1381	CR-2015-1394	CR-2015-1422
CR-2015-1495	CR-2015-1648	CR-2015-1913
CR-2015-1964	CR-2015-2611	CR-2015-2628
CR-2015-5473	CR-2015-5518*	

Work Order

52622914	00404116	00404107	00365133
00406937	00406962	00408682	

Miscellaneous
Pilgrim Technical Specifications
FSAR Section 5.3.3.4, Standby Gas Treatment System
FSAR Section 10.8, Fire Protection System
ESOMS Narrative Log
ESOMS LCO Tracking Program
Fire Protection Program Health Report

Section 1R05: Fire Protection

Procedures
5.5.2, Special Fire Procedure, Revision 52

Condition Reports (*NRC Identified)
CR-2015-3970*

Drawings
A316 Sh 1, Reactor & Turbine Building Floor Plan El. -17'-6" & 6'-0" Fire Barrier System, Revision 6
A316 Sh 2, Reactor & Turbine Building Floor Plan El. -17'-6" & 6'-0" Fire Barrier System, Revision E1
A317 Sh 1, Reactor & Turbine Building Floor Plan El. 23' – 0" Fire Barrier System, Revision E9
A320 Sh 1, Reactor Building Plans El. 117'-0", 101'-0", 91'-3", 74'-3" & Intake Building Plan - Fire Barrier System, Revision E4

Work Order
52505412

Miscellaneous
Fire Hazards Analysis – Fire Area 1.21, Fire Zone 1.21, 'A' RBCCW Pumps/Heat Exchanger Room
Fire Hazards Analysis – Fire Area 1.10, Fire Zone 1.22, 'B' RBCCW Pumps/Heat Exchanger Room
Fire Hazards Analysis – Fire Area 1.10, Fire Zone 1.5, Reactor Core Isolation Cooling Pump Quadrant
Fire Hazards Analysis – Fire Area 1.9, Fire Zone 1.32, Main Steam and Feedwater Tunnel
Fire Hazards Analysis – Fire Area 1.30, Fire Zone 1.30, Drywell
Fire Hazards Analysis – Fire Area 5.2, Fire Zone 5.2, 'B' Train Service Water Pumps Room
Fire Hazards Analysis – Fire Area 5.3, Fire Zone 5.3, 'C' Service Water Pump Room

Section 1R06: Flood Protection Measures

Condition Reports
CR-2002-13064

Calculations
FP51, Expected Maximum Flow from Sprinkler Systems in EDG Rooms

Evaluations
Safety Evaluation 1019, Pipe Supports for Diesel Generator Room Pre-Action Sprinkler Piping
Safety Evaluation 839, Installation of Pre-Action Sprinkler Systems and Hose Stations in the Diesel Generator Rooms
BLE-2251, Flooding from Plant Leakage

Miscellaneous
FSAR Section 8.9, Cable Installation Criteria
Boston Edison Company Memorandum dated 10/4/90, Diesel Generator and Screenhouse Building Flood Protection Check Valves
Specification E536, Maximum Flood Heights in each EDG Room
PMRQ 50078908, 50076539, and 50076538

Section 1R07: Heat Sink Performance

Procedures
3.M.4-98, RBCCW Heat Exchanger Tube, Channel Cover, Channel Shell, and Partition Plate Repair, Revision 24
TP15-004, General Procedure for Eddy Current Testing of Heat Exchanger Tubing, Revision 0

Condition Reports
CR-2015-3702
CR-2015-3779

Work Orders
52517284
00361934

Drawings
M11-26-2 Sh2, RBCCW E-209A Tube Layout as of April 2007, Revision 12
M212 Sh1, P&ID Service Water System, Revision 96
M215 Sh1, P&ID Cooling Water System Reactor Building, Revision 52

Miscellaneous
M591, SSW & RBCCW Safety-Related Piping & Heat Exchanger Inspection, Maintenance, and Test Requirements in Response to Generic Letter 89-13, Revision E7

Section 1R08: Inservice Inspection

Procedures
CEP-NDE-0641, Liquid Penetrant Examination (PT) for ASME Section XI, Revision 7
CEP-NDE-0731, Magnetic Particle Examination (MT) for ASME Section XI, Revision 3

CEP-NDE-0901, VT-1 Examination (Visual), Revision 4
CEP-NDE-0903, VT-3 Examination (Visual), Revision 5
CEP-NDE-0477, Manual Ultrasonic Examination of Austenitic and
Ferritic Vessels Not Greater than 2” in Thickness (ASME XI), Revision 4
GEH-UT-247, Addenda 0 Phased Array Ultrasonic Examination of Dissimilar Metal Welds, Revision 3
GEH-PDI-UT-2, Generic Procedure for the UT Examination of Austenitic Pipe Welds, Revision 7
SEP-CISI-PNPS-001, ASME Code Visual Examination of Primary Containment, Revision 0

Condition Reports
CR-2015-3525
CR-2011-2210

Miscellaneous
INR P1R20 IVVI 15-02 Jet Pump 05 WD-1 Indication Notification Report
INR P1R20 IVVI 15-01 Jet Pump 11 WD-1 Indication Notification Report
INR P1R20 IVVI 15-03 Jet Pump 3, 4 IN-5 Bolting
INR P1R20 IVVI 15-04R1 Steam Dryer Tie Bar TB08 CD C3
IWE-GVWD-01 WO 378055 Walk around and IWE visual exam of Containment Elev 23’
Mercury Exclusion Certificate-Dry magnetic inspection powder
Spot-check Penetrant –Sulfur and halogen free per ASME Section V
NDE Examiner Certification Review
Visual Acuity Examination Record
Certification Number 0904 for Manual Ultrasonic of Pipe to Elbow RHR Austenitic Piping

Section 1R11: Licensed Operator Requalification Program

Miscellaneous
Module 0-RQ-06-02-51, Scenario 15; Momentary loss of Y-1, Loss of Bus A6, and Station Blackout; Revision 1
Module 0-RQ-06-02-95, Scenario 3; EOP-4, Fuel Failure, RWCU Leak with Max Safe Operating Radiation in 2 Areas, EOP-5 Entry; Revision 2

Section 1R12: Maintenance Effectiveness

Procedures
EN-DC-203, Maintenance Rule Program, Revision 3
EN-DC-205, Maintenance Rule Monitoring, Revision 5
2.2.25, Fire Water Supply System, Revision 59
2.5.2.71, Radwaste Collection System, Revision 36
8.M.2-5, Drywell Drain Sump Integrator, Revision 9
8.M.3-17.3, Drywell Equipment and Floor Sump Level Switch Calibration, Revision 5

Condition Reports (*NRC Identified)

CR-2013-3507	CR-2013-7369	CR-2013-0227
CR-2013-0057	CR-2015-2672	CR-2014-2544
CR-2014-2654	CR-2014-3239	CR-2014-3331
CR-2014-3549	CR-2014-4268	CR-2014-4358
CR-2014-4925	CR-2014-5022	CR-2014-5445
CR-2014-5464	CR-2014-5507	CR-2014-6024
CR-2014-6356	CR-2015-0621	CR-2015-0727

CR-2015-0791	CR-2015-0795	CR-2015-0935
CR-2015-0936	CR-2015-1381	CR-2015-1394
CR-2015-1422	CR-2015-1495	CR-2015-1648
CR-2015-1913	CR-2015-1964	CR-2015-2611
CR-2015-2628	CR-2015-5473	CR-2015-5518*

Work Orders

52610735	52313906	00384204	00385788
00389225	00389225	00404116	00404107
00365133	00406937	00406962	00408682
00408765	52610381	52617220	52604684

Drawing
M232, P&ID Radwaste Collection System, Revision 38
M218 Sh2, P&ID Fire Protection System, Revision 46

Miscellaneous
Technical Specifications
Maintenance Rule Basis Document - Radwaste Collection System
Radwaste System Health Reports
FSAR Section 4.10 Nuclear System Leakage Rate Limits
FSAR Section 10.13, Equipment and Floor Drainage Systems
FSAR Section 10.8, Fire Protection System
Maintenance Rule Basis Document for the Fire Protection System
Fire Protection Program Health Report
ESOMS Narrative Logs

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
EN-WM-104, On Line Risk Assessment, Revision 10
1.5.22, Risk Assessment Process, Revisions 25
3.M.1-45, Outage Shutdown Risk Assessment, Revision 17
8.M.1-32.6, Analog Trip System Trip Unit Calibration Cabinet C2233A Section B – Critical Maintenance, Revision 36
TP15-010, RFO 20 Compensatory Measures, Revision 0
3.M.3-33, 345kV Startup Transformer Calibration and Functional Relay Testing – Critical Maintenance, Revision 32
5.3.25.1, Transient Response Hardcards for Operating Crews, Revision 17
EN-FAP-OM-012, Prompt Investigation, Notifications and Duty Manger Responsibilities, Revision 7

Condition Reports (*NRC Identified)
CR-2015-4115

Work Orders
52499608

Miscellaneous
Activity Risk Compensatory Measures
Equipment Out-of-Service Risk Assessment Tool

ESOMS Narrative Log
ESMS Clearance Module
Protected Equipment List
Online risk assessment for the week of 3/29/15
Online risk assessment for the week of 4/12/15
Online risk assessment for the week of 6/7/15
Outage Risk Assessment Review Checklists
Refueling Outage Schedule

Section 1R15: Operability Determinations and Functionality Assessments

Procedures
2.5.2.71, Radwaste Collection System, Revision 36
8.M.2-5, Drywell Drain Sump Integrator, Revision 9
8.M.3-17.3, Drywell Equipment and Floor Sump Level Switch Calibration, Revision 5
EN-OP-104, Operability Determination Process, Revision 8
TP15-010, RFO20 Compensatory Measures, Revision 0
8.A.2, Drywell to Suppression Chamber Vacuum Breaker Leakage Rate Test, Revision 37

Condition Report

CR-2015-2672	CR-2015-2697	CR-2015-2768
CR-2015-4025	CR-2015-5586	

Work Orders
52610735
52313906

Drawing
M232, P&ID Radwaste Collection System, Revision 38

Miscellaneous
Technical Specifications
FSAR Section 4.10 Nuclear System Leakage Rate Limits, Revision 28
FSAR Section 10.13, Equipment and Floor Drainage Systems, Revision 26
FSAR Section 10.7, Salt Service Water System, Revision 29
FSAR Section 1.2, Definitions, Revision 29
Plant Health Committee 2014 Q1 Update, February 4, 2014
SEP-ISI-PNPS-001, ASME B&PV Code Section XI Fourth Ten-Year Inspection Interval Inservice Inspection (ISI) Program Plan July 1, 2005 to June 30, 2015
Specification M591, SSW & RBCCW Safety-Related Piping & Heat Exchanger Inspection, Maintenance, & Test Requirements in Response to Generic Letter 89-13, Revision E7
E536, Environmental Parameters for Use in the Environmental Qualification of Electrical Equipment per 10 CFR 50.49, Revision 12
EQML, Environmental Qualification Master List, Revision 51

Section 1R19: Post-Maintenance Testing

Procedures
2.2.8, Standby AC Power Source (Diesel Generators), Revision 107
8.9.1.2, Diesel Air Start and Turbo Assist System Leak Test, Revision 14
8.5.2.3, LPCI and Containment Cooling Motor Operated Valve Operability Test, Revision 52,

8.I.32, Obtaining Field Stroke Time Data for Establishing Inservice Test (IST) and Appendix B Test (ABT) Programs Power Operated Valve Acceptance Criteria, Revision 8
3.M.4-12.2, Salt Service Water Pumps – Routine Maintenance, Revision 64
8.5.3.2.1, Salt Service Water Pump Quarterly and Biennial (Comprehensive) Operability and Valve Operability Tests, Revision 31
3.M.4-8, Main Steam Isolation Valve Maintenance – Critical Maintenance, Revision 50
3.M.4-8.1, Main Steam Isolation Valve Preventive maintenance – Critical Maintenance, Revision 20
8.7.1.6, Local Leak Rate Testing of the Main Steam Isolation Valves, Revision 30
8.I.11.21, Main Steam Isolation Valve Cold Shutdown Operability, Revision 2
EN-DC-136, Temporary Modifications, Revision 11
EN-MA-125, Troubleshooting Control of Maintenance Activities, Revision 17

Condition Reports (*NRC Identified)

CR-2015-1717	CR-2015-1792	CR-2015-1760
CR-2015-1818*	CR-2015-4208	CR-2015-4222
CR-2015-4206	CR-2015-4865	CR-2015-4415
CR-2015-4380		

Maintenance Orders/Work Orders

52252986	52252987	52565550	52565551
52607011	52480795	00353644	52523432
00397726	00413258	51552016	004134

Drawings
M219, P&ID Diesel Generator Air Start System, Revision 24
M259, P&ID Diesel Generator Turbo Air Assist System, Revision E10
E16A2-8, Elementary & Connection Diagram Vital MG Set, Revision 6

Miscellaneous
FSAR Section 8.5 Standby AC Power Source, Revision 26
FSAR Section 5.2 Primary Containment System, Revision 27
FSAR Section 4.6 Main Steam Isolation Valves, Revision 29

Section 1R20: Refueling and Other Outage Activities

Procedures
EN-OM-123, Fatigue Management Program, Revision 11
EN-RE-215, Reactivity Maneuver Plan, Revision 5
2.1.5, Controlled Shutdown from Power, Revision 125
2.1.7, Vessel Heatup and Cooldown, Revision 54
2.1.31, Rod Worth Minimizer Operability Test, Revision 16
9.29, Control Rod Sequence Development and Programming of the Rod Worth Minimizer, Revision 35
3.M.4-9, Inspection of the Drywell and Suppression Chamber, Revision 20
2.2.85, Fuel Pool Cooling and Filtering System, Revision 89
2.1.8.5, Reactor Vessel Pressurization and Temperature Control for Class 1 System Leakage Test, Revision 33
2.1.1, Startup from Shutdown, Revision 191
2.1.4, Approach to Critical and Plant Heatup, Revision 36
2.1.14, Station Power Changes, Revision 113

Drawings
M209, P&ID Condensate & Demineralized Water Storage & Transfer Systems, Revision 67
M231, P&ID Fuel Pool Cooling and Demineralizer System, Revision 44

Condition Reports (*NRC Identified)

CR-2015-3289	CR-2015-3436	CR-2015-4530*
CR-2015-5106	CR-2015-5341	CR-2015-6024*

Miscellaneous
Reactivity Maneuver Plan RMP-PNP-20-36
Reactivity Maneuver Plan RMP-PNP-21-01
ESOMS Personnel Qualifications & Scheduling database
Pilgrim Security Access Database

Section 1R22: Surveillance Testing

Procedures
2.2.6, 4160V AC System, Revision 55
8.7.3, Secondary Containment Leak Rate Test, Revision 63
8.9.16.2, Manual Start and Loading of Station Blackout Diesel Generator via Safety Bus A5 or A6, Revision 10
8.7.1.6, Local Leak Rate Testing of the Main Steam Isolation Valves, Revision 30
8.M.3-1, Special Test for Automatic ECCS Load Sequencing of Diesels and Shutdown Transformer with Simulated Loss of Offsite Power and Special Shutdown Transformer Load Test – Critical Maintenance, Revision 59
8.M.1-31, SDV Vent and Drain Timing, Revision 28
8.5.6.2, Special Test for ADS System Manual Opening of Relief Valves, Revision 38
8.5.4.1, High Pressure Coolant Injection (HPCI) System Pump and Valve Quarterly and Biennial Comprehensive Operability, Revision 116

Condition Reports

CR-2015-2957	CR-2015-2937	CR-2015-3601
CR-2015-3571	CR-2015-3574	

Work Order

52482622	52511539	00374498	00374497
00374496	00374495	00374494	00374493
00374491	00351288	52514362	52518879

Drawings
M283, Secondary Containment isolation Control Diagram, Revision E9

Miscellaneous
FSAR Section 5.3, Secondary Containment System, Revision 21
FSAR Section 8.10, Blackout AC Power Source, Revision 23
FSAR Section 5.2 Primary Containment System, Revision 27
FSAR Section 4.6 Main Steam Isolation Valves, Revision 29
INI-276, Uncertainty Calculation Reactor Building DP Manometers

Section 2RS1: Access Control to Radiologically Significant Areas

Procedures
EN-RP-109, Audit Process, Revision 28
EN-RP-100, Radiation Worker Expectations, Revision 9
EN-RP-101, Access Controls for Radiologically Controlled Areas, Revision 10
EN-RP-102, Radiological Control, Revision 4
EN-RP-104, Personnel Contamination Events, Revision 7
EN-RP-106, Radiological Survey Documentation, Revision 5
EN-RP-108, Radiation Protection Posting, Revision 16
EN-RP-122, Alpha Monitoring, Revision 8
EN-RP-204, Special Monitoring Requirements, Revision 6
PNPS 6.1-220, Radiological Controls for High Risk Evolutions, Revision 15

Condition Reports
CR-2015-03925

Documents
LO-PNPLO-2015-00132, Contamination Control, Self-Assessment, January 6, 2015
S. Brewer to A. Zelie, PNPS Radiological Support Group 4th Quarter 2014 Self-Assessment Report, March 3, 2015
PNPS-RWP-2015-0490, Post Drain Down Cavity Decon, November14, 2014
PNPS-RWP-2015-0509, Exchange CRDs and Support Activities, April 27, 2015
PNPS-RWP-2015-530, Work Inside A & B Condensers – Dog Bone Replacement, May 5, 2015
PNPS-RWP-2015-538, Moisture Separator Repairs RFO-20, April 29, 2015
PNPS-RWP-2015-542, Torus De-sludge RFO-20, April 23, 2015
PNPS-RWP-2015-481, Scaffolding RFO-20, October 13, 2014

Section 2RS2: Occupational ALARA Planning and Controls

Procedures
EN-RP-110-05, ALARA Planning and Controls, Revision 2
EN-RP-110-04, Radiation Protection Risk Assessment Process, Revision 4
PNPS Procedure 6.1-220, Radiological Controls for High Risk Evolutions, Revision 15
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LIST OF ACRONYMS

10 CFR	Title 10 of the *Code of Federal Regulations*
AC	alternating current
ACE	apparent cause evaluation
ADS	automatic depressurization system
AFPC	Augmented Fuel Pool Cooling
ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
B&PV	boiler and pressure vessel
CAP	Corrective Action Program
CR	condition report
GPM	gallons per minute
HPCI	high pressure coolant injection
HRA	high radiation area
IMC	inspection manual chapter
ISI	in service inspection
kV	kilovolt
LCO	limiting condition for operation
LER	licensee event report
LHRA	locked high radiation area
LPCI	low pressure coolant injection
MR	maintenance rule
MT	magnetic particle test
NCV	non-cited violation
NDE	nondestructive examination
NRC	Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
PT	liquid penetrant test
RAM-QC	radioactive material-quality of concern
RBCCW	reactor building closed cooling water
RFO	refueling outage
RG	regulatory guide
RHR	residual heat removal
SBO	station blackout
SDT	shutdown transformer
SGT	standby gas treatment
SR	surveillance requirement
SRV	safety relief valve
SSC	structure, system, or component
TS	technical specification
UFSAR	Updated Final Safety Analysis Report
UT	ultrasonic test
VHRA	very high radiation area
WO	work order