Entergy

Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
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Plymouth, MA 02360

Robert G. Smith, P.E.
Site Vice President

April 21, 2011

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No.: 50-293
License No.: DPR-35

Licensee Event Report 2011-001-00, Technical Specification (TS) Required Shutdown – RBCCW "B" Declared Inoperable

Licensee Event Report 2011-002-00, Reactor Scram During A Planned Reactor Cool-Down with All Control Rods Fully Inserted

LETTER NUMBER: 2.11.029

Dear Sir or Madam:

The enclosed Licensee Event Reports (LERs) 2011-001-00, "Technical Specification (TS) Required Shutdown – RBCCW "B" Declared Inoperable" and 2011-002-00 "Reactor Scram During A Planned Reactor Cool-Down with All Control Rods Fully Inserted" are submitted in accordance with 10 CFR 50.73.

This letter contains no commitments.

Please do not hesitate to contact Mr. Joseph R. Lynch, (508) 830-8403, if there are any questions regarding this submittal.

Sincerely,

for

Robert G. Smith

RMB/rmb

Attachments: 1. Licensee Event Report 2011-001-00, Technical Specification (TS) Required Shutdown – RBCCW "B" Declared Inoperable (6 Pages)
2. Licensee Event Report 2011-002-00, Reactor Scram During A Planned Reactor Cool-Down with All Control Rods Fully Inserted (5 Pages)

IE22
NRR

cc:

Mr. William M. Dean
Regional Administrator, Region 1
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USNRC Senior Resident Inspector
Pilgrim Nuclear Power Station

Attachment 1
Letter Number 2.10.029

Licensee Event Report 2011-001-00,
Technical Specification (TS) Required Shutdown – RBCCW "B" Declared Inoperable

(6 pages)

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013
LICENSEE EVENT REPORT (LER)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Service Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Technical Specification (TS) Required Shutdown – RBCCW 'B' Declared inoperable

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER 05000
02	20	2011	2011	001	00	04	20	2011	FACILITY NAME N/A	DOCKET NUMBER 05000

9. OPERATING MODE: Run

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☒ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (*Include Area Code*)
Mr. Joseph R. Lynch, Licensing Manager	(508)-830-8403

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	CC	HX	E270	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ Yes (If yes, complete 15. *EXPECTED SUBMISSION DATE*) ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 0055 hours on Sunday, February 20, 2011, the Pilgrim Nuclear Power Station (PNPS) commenced a controlled shutdown of the reactor due to the 'B' train of Reactor Building Closed Cooling Water (RBCCW) being declared inoperable and expected to exceed its 72-hour Limiting Condition for Operability (LCO) as required by TS prior to return to operable status.

With the plant operating at 100% power, leakage of Salt Service Water (SSW) was detected in the RBCCW system due to high chloride levels and increased inventory in the system. An investigation into the event determined that the source of the SSW was isolated to the 'B' RBCCW heat exchanger which is designed to cool RBCCW under normal and post-accident conditions. The quantity of the leakage was determined to exceed the design limits established to ensure post-accident operation of the system and the 'B' train of RBCCW was subsequently declared inoperable.

The leak detection and repair activities identified a single tube leak resulting from an improperly modified tube sleeve (shortened and incorrect bevel) which accelerated wear on the parent tube.

This event had no impact on the health and/or safety of the public.

NARRATIVE

EVENT DESCRIPTION:

At 0055 hours on Sunday, February 20, 2011, the Pilgrim Nuclear Power Station (PNPS) commenced a controlled shutdown of the reactor due to the 'B' train of Reactor Building Closed Cooling Water (RBCCW) being declared inoperable and expected to exceed its 72-hour Limiting Condition for Operability (LCO) as required by TS prior to return to operable status.

With the plant operating at 100% power, leakage of Salt Service Water (SSW) was detected in the RBCCW system due to high chloride levels and increased inventory in the system. An investigation into the event determined that the source of the SSW was isolated to the 'B' RBCCW heat exchanger which is designed to cool RBCCW under normal and post-accident conditions. The quantity of the leakage was determined to exceed the design limits established to ensure post-accident operation of the system and the 'B' train of RBCCW was subsequently declared inoperable.

The leak detection and repair activities identified a single tube leak resulting from an improperly modified tube sleeve (shortened and incorrect bevel). The modified sleeve was installed in a 2005 maintenance outage and over time accelerated wear on the parent tube.

BACKGROUND:

The Reactor Building Closed Cooling Water (RBCCW) System provides cooling to the Core Standby Cooling System (CSCS) components and provides a heat sink for the Residual Heat Removal (RHR) System heat exchangers. The system also provides required cooling to the equipment located in the Reactor Building during normal planned station operations, and to provide a barrier between the primary system and the Salt Service Water (SSW) System.

The RBCCW System consists of two independent closed loops for redundancy during accident conditions. Each loop has three centrifugal pumps and takes suction from the associated RBCCW heat exchanger. A 500 gallon head tank for each loop is located at the highest point in the system and accommodates system volume changes, maintains static pressure in the loop, detects gross leaks in the system, and provides a means for adding makeup water. The two loops can be cross-tied through two 12-inch cross-tie headers using four valves. The cross-tie valves are normally closed.

During plant operations, the RBCCW system also functions as an intermediate barrier between system equipment and the SSW system. The RBCCW loop pressure is normally higher than the salt service water system pressure preventing salt water contamination of the RBCCW system. Detectors in the RBCCW system continuously monitor radioactivity levels.

The RBCCW heat exchangers were placed in operation in approximately 1971. As a result of the station's eddy current testing program tube sleeves (also called inserts, shields or ferrules) were installed in the mid 1980's in both RBCCW and TBCCW heat exchangers. These sleeves were made of the same material as the tubes, 90-10 Copper Nickel.

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EVENT ANALYSIS:

At 0055 hours on Sunday, February 20, 2011, the Pilgrim Nuclear Power Station (PNPS) commenced a controlled shutdown of the reactor due to the 'B' train of Reactor Building Closed Cooling Water (RBCCW) being declared inoperable and expected to exceed its 72-hour Limiting Condition for Operability (LCO) as required by TS prior to return to operable status.

With the plant operating at 100% power, leakage of Salt Service Water (SSW) was detected in the RBCCW system due to high chloride levels and increased inventory in the system. An investigation into the event determined that the source of the SSW was isolated to the 'B' RBCCW heat exchanger which is designed to cool RBCCW under normal and post-accident conditions. The quantity of the leakage was determined to exceed the design limits established to ensure post-accident operation of the system and the 'B' train of RBCCW was subsequently declared inoperable. A 10 CFR 50.72 report was required because the subsystem would not have been restored to operable status prior to exceeding the 72 hour TS LCO action statement for one RBCCW subsystem inoperable as defined in TS 3.5.B.3.B. As a result, a Licensee Event Report (LER) per 10 CFR 50.73 is required.

The leak detection and repair activities identified a single tube leak resulting from an improperly modified tube sleeve (shortened and incorrect bevel) which accelerated wear on the parent tube. An extent of condition was performed on the affected heat exchanger and no additional repairs were necessary.

CAUSE OF EVENT:

The direct cause of the SSW leak into the "B" loop RBCCW system was a tube leak in the RBCCW 'B' heat exchanger (E-209B) related to the installation of an undocumented and unauthorized field modified (shortened) inlet end sleeve resulting in accelerated wear on the parent tube.

EXTENT OF CONDITION;

All remaining sleeves in E-209B were inspected and verified to be the correct length (8"). The other components that may have modified sleeves installed were the "A" RBCCW heat exchanger (E-209A), and the "A" and "B" TBCCW heat exchangers (E-122A & B, respectively). The existing 8" sleeves are planned to be replaced with 10" long sleeves during the next scheduled maintenance for all of these heat exchangers.

1. E-209A was last inspected in RFO 16 (April 2007) and all sleeves were replaced. There were no indication of any leakage concerns and the heat exchanger is currently scheduled to be inspected in RFO 18 (April 2011) and have all the sleeves replaced with 10" long sleeves.

2. E-122A was last inspected in February 2009 and all 8" long sleeves were replaced at that time. There were 2 tube leaks downstream of inlet end sleeves in December 2010 that were attributed to end step erosion due to high flow (velocity) operation. The inlet end sleeves removed from the leaking tubes were the correct design (8" long). This heat exchanger is scheduled to have all the sleeves replaced with 10" long sleeves in 2011.

3. E-122B was last inspected in March 2010 and all of the sleeves were replaced in kind (8") at that time. The next scheduled inspection for E-122B is March 2014. After completion of sleeve replacement in E-122A, the scheduled inspection for E-122B will be reevaluated and revised as required.

FAILED COMPONENT IDENTIFICATION:

The following EIIS codes are applicable to this report:

<u>COMPONENTS</u>	<u>CODES</u>
Heat Exchanger	HX
<u>SYSTEMS</u>	<u>CODES</u>
Closed / Component Cooling Water Systems (RBCCW)	CC

CORRECTIVE ACTIONS:

<u>Completed Actions:</u>

1. Verified that 100% of the remaining sleeves installed in E-209B heat exchanger are the correct length.
2. Performed the following maintenance activities on E-209B heat exchanger:
    a. Open, pressure test and identify leaking tube(s)
    b. Inspect other tubes as required
    c. Repair (plug) leaking tube(s)
    d. Restore E-209B to operation
3. Revised and issued procedures 3.M.4-98 (RBCCW Heat Exchanger maintenance/repair) and 3.M.4-99 (TBCCW maintenance/repair) to include language to preclude recurrence. **Note: If difficulties are encountered when installing sleeves, notify maintenance management. Sleeves should not be modified without an approved design change. Ref CR-PNP-2011-0721.**

<u>Open Actions:</u>

1. Create a signature for verification that the heat exchanger maintenance/repair procedure was completed as written and that a Condition Report (CR) would be written if it could not be done as written.
2. In addition to scheduled inspections, perform maintenance on E-209A including the installation of replacement tube sleeves during RFO 18.

ASSESSMENT OF SAFETY CONSEQUENCES:

This LER is submitted pursuant to the requirements of 50.73(a)(2)(i)(A) because a Technical Specification required shutdown was completed.

The event occurred during normal power operation while at 100% power with the mode switch in the “RUN” position. The reactor vessel pressure was approximately 1030 psig with reactor water temperature at saturation temperature for that pressure.

The Core Standby Cooling Systems (CSCS) consist of the High Pressure Coolant Injection (HPCI) System, Automatic Depressurization System (ADS), Core Spray (CS) System, and the Residual Heat Removal (RHR) System in the Low Pressure Core Coolant Injection (LPCI) mode. Although not part of the CSCS, the Reactor Core Isolation Cooling (RCIC) System is capable of providing water to the reactor vessel for high pressure core cooling, similar to the HPCI System. These systems were operable when the RBCCW Train ‘B’ was declared inoperable.

The RBCCW System provides cooling to the CSCS System components and provides a heat sink for the RHR System heat exchangers. The system also provides required cooling to the equipment located in the Reactor Building during normal planned station operations, and provides a barrier between the primary system and the Salt Service Water (SSW) System.

The RBCCW System consists of two independent closed loops. Each loop has three centrifugal pumps and takes suction from the RBCCW heat exchanger. A 500 gallon head tank is located at the highest point in the system and accommodates system volume changes, maintains static pressure in the loop, detects gross leaks in the system, and provides a means for adding makeup water. Head tank level is monitored and will alarm in the main control room if a level deviation exists. The system is designed with sufficient redundancy so that no single system component failure can prevent the system from performing its safety objective.

The heat exchanger leakage in the RBCCW ‘B’ Train was being closely monitored upon discovery of the high chlorides in the system along with the increase in head tank. The ‘A’ RBCCW heat exchanger was operable prior to and through the event.

The leak posed no threat to public health and safety. On initial identification of a leak, the plant developed an Operational Decision Making Issue (ODMI) "Non-Accident RBCCW Surge Tank T-201B Monitoring" to track and trend leakage. The trigger points identified in the ODMI were subsequently reached and the plant entered an LCO and shutdown. During the shutdown the RBCCW heat exchanger was repaired, determined to be operable and returned to service

SIMILAR EVENTS:

A review was conducted of Pilgrim Station LERs which involved either failure of an RBCCW subsystem or failure of the heat exchanger The following LER addressed similar concerns:

LER 96-008-00 describes an event where a leak was discovered on the RBCCW “B” Train heat exchanger and the condition resulted in declaring the subsystem inoperable. The condition resulted in entry into TS LCO 3.5.B.3.B and subsequent plant shutdown.

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		YEAR	SEQUENTIAL NUMBER	REV NO.	
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REFERENCES:

CR-PNP-2011-00721
INPO Failure Report Number 528
PNPS Procedure No. 3.M.4-98
PNPS Procedure No. 3.M.4-99

Attachment 2
Letter Number 2.10.029

Licensee Event Report 2011-002-00,
Reactor Scram During A Planned Reactor Cool-Down with All Control Rods Fully Inserted

(5 pages)

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Reactor Scram During A Planned Reactor Cool-Down with All Control Rods Fully Inserted

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER 05000
2	20	2011	2011-002-00			4	20	2011	FACILITY NAME N/A	DOCKET NUMBER 05000

9. OPERATING MODE: Startup

10. POWER LEVEL: 0

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

- [ ] 20.2201(b)
- [ ] 20.2201(d)
- [ ] 20.2203(a)(1)
- [ ] 20.2203(a)(2)(i)
- [ ] 20.2203(a)(2)(ii)
- [ ] 20.2203(a)(2)(iii)
- [ ] 20.2203(a)(2)(iv)
- [ ] 20.2203(a)(2)(v)
- [ ] 20.2203(a)(2)(vi)
- [ ] 20.2203(a)(3)(i)
- [ ] 20.2203(a)(3)(ii)
- [ ] 20.2203(a)(4)
- [ ] 50.36(c)(1)(i)(A)
- [ ] 50.36(c)(1)(ii)(A)
- [ ] 50.36(c)(2)
- [ ] 50.46(a)(3)(ii)
- [ ] 50.73(a)(2)(i)(A)
- [ ] 50.73(a)(2)(i)(B)
- [ ] 50.73(a)(2)(i)(C)
- [ ] 50.73(a)(2)(ii)(A)
- [ ] 50.73(a)(2)(ii)(B)
- [ ] 50.73(a)(2)(iii)
- [X] 50.73(a)(2)(iv)(A)
- [ ] 50.73(a)(2)(v)(A)
- [ ] 50.73(a)(2)(v)(B)
- [ ] 50.73(a)(2)(v)(C)
- [ ] 50.73(a)(2)(v)(D)
- [ ] 50.73(a)(2)(vii)
- [ ] 50.73(a)(2)(viii)(A)
- [ ] 50.73(a)(2)(viii)(B)
- [ ] 50.73(a)(2)(ix)(A)
- [ ] 50.73(a)(2)(x)
- [ ] 73.71(a)(4)
- [ ] 73.71(a)(5)
- [ ] OTHER Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Joseph R. Lynch, Licensing Manager	(508)-830-8403

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
[ ] Yes (If yes, complete 15. *EXPECTED SUBMISSION DATE* [X] NO		N/A	N/A	N/A

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On Sunday, February 20, 2011 at 1034 EST, with the reactor shutdown and all control rods fully inserted a valid Reactor Protection System (RPS) low reactor water level initiation signal (+12 inches) was received. The RPS actuation signal resulted in a reactor scram and actuation of Primary Containment Isolation System (PCIS) Group II (Drywell) isolation, Group VI (RWCU) isolation and a Reactor Building Isolation System (RBIS) actuation. At the time of the event, a controlled reactor shutdown and cooldown was in progress. The Reactor Mode Selector Switch was in "Startup" and the low reactor water level actuation signal was the result of reactor water level control difficulties during the cool-down using the Mechanical Pressure Regulator (MPR). Reactor water level was immediately restored, the isolations (Group II and VI) were reset; and the RPS signal was reset at 1135 EST. All systems operated as expected, in accordance with design.

Corrective actions taken included the revision of the reactor heat-up / cool-down procedure to incorporate lessons learned and to identify the Bypass Valve Opening Jack (BVOJ) as the preferred method for executing a reactor pressure vessel cool-down. Corrective actions planned include the performing of an analysis of MPR/RPV and level response during plant cool-down at the plant simulator and evaluate results for disposition. This event had no impact on the health and/ or safety of the public.

NARRATIVE

EVENT DESCRIPTION:

On 2/20/2011, a planned reactor shutdown/ cooldown was being performed in accordance with PNPS Procedures 2.1.5, Controlled Shutdown from Power and 2.1.7, Vessel Heatup and Cooldown to address leakage within the Reactor Building Closed Cooling Water (RBCCW) Loop 'B' heat exchanger (Reference LER 2011-001-00). During cool-down, with the startup feedwater regulating valve and reactor water cleanup (RWCU) letdown in service, Pilgrim experienced a reactor scram signal, Group II isolation, Group VI isolation and RBIS initiation on low reactor water level (reactor water level reached +10.4 inches - scram set-point is +12 inches). All control rods were fully inserted at the time of the scram. The low reactor water level was the result of reactor water level control difficulties experienced while performing a reactor cool-down using the Mechanical Pressure Regulator (MPR). This method was selected following successful performance in the simulator during Just-In-Time (JIT) training. The use of the MPR was one of two procedurally allowed options for plant cool-down, as the second method being the use of the Bypass Valve Opening Jack (BVOJ).

BACKGROUND:

During steady state and dynamic plant conditions, reactor pressure is maintained by the Mechanical Hydraulic Control (MHC) System. During plant cooldown, one of two mechanisms can be utilized to adjust main steam line, and therefore reactor pressure limiting the rate of temperature reduction to that allowed by Technical Specifications. The Mechanical Pressure Regulator (MPR) adjusts the control bypass valve (BPV) positions to control reactor pressure at an established setpoint. The BVOJ is a motor actuated linkage that can be used to directly open the bypass valves.

EVENT ANALYSIS:

During the event, the MPR was controlling reactor pressure by opening and closing the #1 BPV; the BPV movements caused reactor water level oscillations. This was due to the relatively coarse control nature of this cool-down method. Review following the event with several experienced personnel identified that the BVOJ provides a more fluid depressurization producing vessel level shrink of a much smaller magnitude than the MPR. In this case, the on-off control of the MPR resulted in the "at the controls" (ATC) operator taking action to significantly reduce the feedwater flow rate to prevent a high water level condition.

The vessel level shrink coupled with reduced feedwater flow to account for the reduced vessel inventory resulted in reactor water level lowering below the RPS actuation/RBIS and PCIS isolation setpoint of +12 inches reactor water level. The cool-down operation was stopped and operations management performed a stand-down with the operating crew.

Just In Time (JIT) training had been conducted the previous day. In attendance were the Shift Manager (SM), Control Room Supervisor (CRS), the Administrative Control Room Supervisor, Assistant Control Room Supervisor (ACRS) and two Reactor Operators (ROs). The training included review of PNPS Procedure 2.1.7 Attachment 6, Reactor Pressure Vessel Cooldown Rate Schedule and dynamic implementation in the simulator. The procedure directs, "The cooldown will be accomplished using the MHC System (adjusting the MPR down to 150 psig then the Bypass Valve Opening Jack (BVOJ) for the last 150 psig or using the BVOJ from initial pressure all the way down)." No method was prescribed as preferred by the procedure, training, or collective experience of the participants, so the SM directed that both methods be evaluated by the team. During the JIT training, incremental cooldown steps were performed using both mechanisms (MPR and BVOJ) starting at rated reactor pressure, 1030 psig. Adequate control of depressurization rate and reactor water level was experienced in both cases. A water level swell of approximately 4" occurred when the BPV was opened with a corresponding shrink of the same magnitude when the BPV closed. No adjustment to reactor water level makeup or reject was needed to maintain level within a narrow band. The decision was made by the SM to use the MPR method at the plant.

The cooldown was directed to be executed by the CRS and was initiated by the ATC operator both of who had participated in the JIT training. One of the additional ROs, who had not attended the JIT training, was assigned as the peer check. Because of the effect of normal steam loads, the cooldown was commenced from a lower reactor pressure of approximately 700 psig. The ATC operator began to lower the MPR setpoint in a continuous fashion as directed by the procedure to establish a target pressure to 585 psig. A prompt reactor water level swell of approximately 14" occurred causing the ATC to stop lowering the MPR setpoint and then take action to reduce feed water flow by closing down on the startup regulating valve. When water level began to lower, the operator recommenced lowering pressure using the MPR, and achieved the 585 psig MPR target setpoint. As observed on plant computer traces, level and BPV position cycled a total of 5 times with varying magnitude. The MPR operated by closing the BPV when pressure dropped to the established setpoint causing the cycling of BPV and indicated reactor level. When the BPV closed, vessel shrink combined with a very low feed water flow rate resulted in water level lowering to +12" in less than thirty seconds to a minimum of about 10.5".

A water level of +12" produced the expected plant response (reactor scram signal, primary containment system Group II, VI and reactor building isolations). Following the actuations and initiations, the control room crew verified that all automatic actions had appropriately occurred and took action to re-establish RWCU letdown flow path, reset the scram signal and restore normal ventilation. During the error review meeting, it was clear that the crew understood the fundamental concepts of reactor vessel level swell and shrink during depressurization and stabilization. The impact of nearly securing feedwater flow on indicated level when the MPR setpoint was reached was not fully appreciated by the RO or CRS.

An 8-hour Non-Emergency 10 CFR 50.72 notification was made to the USNRC.

CAUSE OF EVENT:

The Root Cause of the event was a failed opportunity to capture and up-date the reactor cooldown procedure with relevant historical Pilgrim Operating Experience regarding previously attempted cooldown evolutions using the MPR.

Contributing Causes

- The crew did not apply sufficient questioning attitude and stop when unsure, when the magnitude of the initial reactor water level swell exceeded that experienced during Just-In-Time (JIT) training.
- The Simulator did not adequately model the plant's reactor level response during MPR cool-down at these low flow, high temperature conditions. This condition contributed to a high sense of confidence in performing the cooldown with the MPR evolution.

EXTENT OF CONDITION:

A review of station errors and events for the past five years failed to identify any occurrences attributed to deficiencies in any of the following areas: failure to capture pertinent operating experience in procedure or shortfalls in simulator modeling. Accuracy of simulator modeling was additionally evaluated by review of the plant simulator index and simulator design review board meeting minutes. No significant discrepancies were found and existing deficiencies were appropriately prioritized. A review of crew and plant performance where JIT training had been utilized determined that the training was effective in supporting successful performance. Gaps in management oversight were identified in at least two of the occurrences (CR-PNP-2009-0499 and CR-PNP-2009-4036). Management engagement and correction of at risk behaviors is a significant corporate initiative through the use of Entergy Nuclear Platform 3: Set and Continuously Enforce High Standards and Fleet Procedure EN-FAP-OM-001: Leadership Forums for Continuous Improvement. These initiatives are considered sufficient to address extent of problem / condition.

FAILED COMPONENT IDENTIFICATION:

Not applicable.

CORRECTIVE ACTIONS:

Immediate corrective actions taken were to temporarily halt the cool-down operation while operations conducted a stand-down. Plant cool-down was subsequently performed successfully utilizing the BVOJ.

Corrective actions taken included the revision of the reactor heat-up / cool-down procedure to incorporate lessons learned to identify the Bypass Valve Opening Jack (BVOJ) as the preferred method for executing a reactor pressure vessel cool-down.

Corrective actions planned include the performing of an analysis of MPR/RPV and level response during plant cool-down at the plant simulator and evaluate results for disposition.

The corrective actions are being tracked in the Pilgrim Station Corrective Action Program via CR-PNP-2011-00733.

ASSESSMENT OF SAFETY CONSEQUENCES:

The event posed no threat to public health and safety.

A low reactor water level signal (+12") with all control rods fully inserted resulted in a scram signal, PCIS Group II, Group VI and Reactor Building Isolation signals. Following verification that all automatic actions had occurred as expected, the reactor scram and isolation signals were reset, restoring system configurations to their pre-initiation status. The plant remained within the established shutdown risk evaluation of the five key safety functions (Inventory Control, Decay Heat Removal, Power Availability, Reactivity Control and Containment). At its lowest point of +10.5", reactor water level was maintained greater than 10' above top of active fuel. There was no radiological or industrial safety impact. Based on the fact that there was no challenge to nuclear, radiological or industrial safety, the impact on safety was not significant.

SIMILAR EVENTS:

Pilgrim 07/10/2007

Pressure control was oscillating between the EPR, MPR and Bypass Valve Opening Jack. Pressure "Control" and "Not in Control" lights on the EPR, MPR and Bypass Valve Opening Jack were all oscillating.

During a thermal backwash on 7/10/07 while the reactor was at 50% power, the control room received a turbine trip and reactor scram on a low vacuum trip. The reactor was running at roughly 945 psig at the time of the scram. The EPIC traces show the 3 bypass valves open initially, relieve the initial post scram pressure spike and then close as the expected response. Reactor pressure dropped to roughly 840 psig. Within 10 minutes, decay heat was causing the pressure to rise. At this point, reactor pressure only increased to 928 psig when the MPR took control and the # 1 bypass valve began to oscillate. Steam supply pressure only recovered to 922 psig when the bypass valves began to oscillate. The Apparent Cause of the reactor pressure oscillation was a burr on the MPR pilot valve which most likely was caused by a piece of debris within the turbine lube oil system or age related wear to the pilot valve.

REFERENCES:

CR-PNP-2011-0773
PNPS Procedure 2.1.5 Controlled Shutdown from Power
PNPS Procedure 2.1.7 Vessel Heatup and Cooldown