Entergy

Entergy Nuclear Operations, Inc.
600 Rocky Hill Road
Plymouth, MA 02360

Pilgrim Nuclear Power Station

July 22, 2013

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 2013-005-00, Primary Containment Declared Inoperable During HPCI Testing

Licensee Event Report 2013-006-00, HPCI Controller Failure to Achieve Rated Flow while in Auto Mode

LETTER NUMBER: 2.13.058

Dear Sir or Madam:

The enclosed Licensee Event Reports are submitted in accordance with 10 CFR 50.73.

LER 2013-005-00, "Primary Containment Declared Inoperable During HPCI Testing"
LER 2013-006-00, "HPCI Controller Failure to Achieve Rated Flow while in Auto Mode"

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,

DE Noyes

David Noyes
Director, Nuclear Safety Assurance

DN/WGL

Attachment 1: Licensee Event Report 2013-005-00, Primary Containment Declared Inoperable During HPCI Testing (4 pages)

Attachment 2: Licensee Event Report 2013-006-00, HPCI Controller Failure to Achieve Rated Flow while in Auto Mode (4 Pages)

JE22
NRR

cc:

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USNRC Senior Resident Inspector
Pilgrim Nuclear Power Station

Attachment 1
Letter Number 2.13.058

Licensee Event Report 2013-005-00

Primary Containment Declared Inoperable During HPCI Testing

(4 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)

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4. TITLE

Primary Containment Declared Inoperable During HPCI Testing

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	DOCKET NUMBER
									N/A	
05	23	2013	2013	005	00	7	22	2013	FACILITY NAME N/A	DOCKET NUMBER

9. OPERATING MODE

N

10. POWER LEVEL

2%

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)
- ☐ 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

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
A	BJ	N/A	N/A	Y	A	NH	N/A	N/A	Y

14. SUPPLEMENTAL REPORT EXPECTED

☐ Yes (If yes, complete 15. *EXPECTED SUBMISSION DATE*) ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

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

At 0455 hours on Thursday, May 23, 2013, with Pilgrim Station in the Startup/Hot Standby Mode and reactor pressure approximately 550 psig, primary containment was declared inoperable due to a leak in the High Pressure Coolant Injection System (HPCI) turbine exhaust line observed while performing the HPCI system flow test. Power ascension was suspended pending investigation and repair. Pilgrim entered into Technical Specification 3.7.A.2 requiring the plant to be in cold shutdown within 24 hours. All other safety systems functioned as required.

The apparent cause of the leak in the HPCI turbine exhaust line was failure to adequately tighten all of the flange bolting due to unique bolting and flange configuration associated with the new check valve and butterfly valve installation during the refueling outage (RFO)-19. The flange bolting was subsequently re-tightened applying the target torque values for the application. A Type B Local Leak Rate Test was performed on the butterfly valve outlet flange and the measured leak rate met the Technical Specification and 10 CFR 50, Appendix J requirements, and primary containment was declared operable.

This event posed no threat to public health and safety.

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NARRATIVE

BACKGROUND:

During RFO-19, stop check valve 2301-74 in the HPCI turbine exhaust line was replaced by a swing check valve and a new butterfly valve (23-HO-321) was installed at the outlet of 2301-74. The removed valve and the new valves are all flanged-end valves. As part of the plant startup following RFO-19, the HPCI system was tested at 150 psig reactor pressure in accordance with PNPS procedure 8.5.4.3, "High Pressure Coolant Injection Operability Demonstration and Flow Rate Test at 150 psig"; conduct of this procedure was also one of the post work test requirements for the valve replacement modification. During HPCI operation, water leakage was observed from the outlet flange on valve 23-HO-321.

The HPCI turbine exhaust steam is discharged to the torus through two check valves, 2301-45 and 2301-74, both of which are primary containment isolation valves. The HPCI turbine exhaust check valve 2301-45 is located closest to the turbine, and valve 2301-74 is located downstream closest to the torus. These valves are tested locally for leak tightness in accordance with the station's local leak rate testing (LLRT) procedure, PNPS procedure 8.7.1.5, "Local Leal Rate testing of Primary Containment Penetrations, Isolation Valves, and Inspection of Containment", to meet the requirements of 10CFR50 Appendix J. The purpose of the Appendix J is to ensure the integrity of the primary containment to contain any releases of radioactive material to containment inside the primary containment. The piping system flanges are also tested as part of the LLRT program.

Technical Specification 3.7.A. 2 requires primary containment integrity at all times when the reactor is critical or when the reactor coolant temperature is above 212°F and fuel is in the reactor vessel. To assure primary containment integrity, all containment isolation valves must be operable or closed and pressure boundary must remain intact to comply with radiological release limits specified in 10 CFR 100 in the event of a break in the primary coolant system piping.

EVENT DESCRIPTION:

On May 23, 2013, the HPCI system flow test was performed in accordance with PNPS 8.5.4.3 with reactor pressure at 150 psig and Pilgrim Station in the Startup/Hot Standby Mode. During the test, water was observed leaking from the butterfly valve outlet-to-plant piping flange on the HPCI turbine exhaust piping. The water was from condensed steam in the line. Because the leakage indicated there was a leak path past both o-ring seals at this flange joint, Operations made the conservative decision to declare primary containment inoperable. Pilgrim entered into the cold shutdown LCO on May 23, 2013, at 0455 and upon completing the repair, exited the LCO on May 23, 2013, at 1822.

CAUSE OF THE EVENT:

The cause was the failure to provide work package instructions necessary to adequately tighten all of the bolting associated with the affected HPCI turbine exhaust piping flange joint. This was due to the lack of understanding of the joint bolting configuration that 4 of the studs were threaded into each side of the butterfly valve body, which is different from the 16 studs that pass through the butterfly valve flange bolt holes and are captured by nuts at the adjacent check valve and plant piping flanges.

There were no component failures.

CORRECTIVE ACTIONS:

The following corrective actions were implemented:

Once appropriate instructions were provided, the flange bolting was tightened to the target torque value of 700 ft-lb.

The HPCI system was subsequently operated in accordance with PNPS 8.5.4.3 with reactor pressure at approximately 550 psig. A Type B LLRT test was performed on the butterfly valve outlet flange and the measured leak rate met the requirements of PNPS procedure 8.7.1.5, 10CFR50 Appendix J, and the Technical Specifications. The primary containment was then declared operable.

Additional corrections are captured in the corrective action program under Condition Report, CR-PNP-2013-04262.

ASSESSMENT OF SAFETY CONSEQUENCES:

This condition posed no threat to the public health and safety.

The event occurred during power ascension from RFO-19. Core Thermal Power was at approximately 2% and reactor pressure was approximately 550 psig.

The safety objective of Primary Containment is to limit the release of fission products in the event of a design basis accident so that off-site doses would not exceed the requirements of 10 CFR Part 100 and to prevent excessive fuel cladding temperatures. Primary containment consists of the drywell and the pressure suppression chamber (torus). The torus provides the water supply for the Core Standby Cooling Systems.

The HPCI exhaust piping leak identified in this report is located in the HPCI Room. The HPCI Room is located outside of primary containment but inside the Reactor Building or Secondary Containment. Secondary Containment is provided to minimize ground level release resulting from potential leaks from primary containment.

The HPCI exhaust line connects to the Torus and is submerged below water level in the Torus. A vacuum breaker line is provided to prevent condensing steam in the exhaust line from drawing a vacuum and drawing water into to exhaust line which would result in water hammer. There are two motor operated valves in the vacuum breaker line.

During the time period that the primary containment barrier was degraded, Secondary Containment was Operable as was the motor operated valves in the vacuum breaker, which were able to isolate the exhaust line leak. In addition, the ADS, CS, RHR, and RCIC systems were either operable or available. These systems and components provided capability to isolate the identified leak path and supply makeup water to the vessel to ensure adequate core cooling.

The leak was repaired and there was no long term safety significance associated with the event.

REPORTABILITY:

This occurrence was reported to the USNRC in accordance with 10 CFR 50.72(b)(3)(ii)(A) and 10 CFR 50.72(b)(3)(v) (C) and (D) as documented in EN# 49061.

PREVIOUS OCCURRENCES:

There were no previous events related to the HPCI turbine exhaust piping flange leakage.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES:

The EIIS codes for Components and Systems referenced in this report are as follows:

COMPONENTS	CODES
Pipe Fittings	PSF
Valve, Isolation	ISV
SYSTEMS	
High Pressure Coolant Injection (HPCI)	BJ
Reactor Containment Building (BWR Primary Containment	NH

REFERENCES:

Condition Report, CR-PNP-2013-04262, HPCI Turbine Exhaust Piping Flange Leak

Attachment 2
Letter Number 2.13.058

Licensee Event Report 2013-006-00

HPCI Controller Failure to Achieve Rated Flow while in Auto Mode

(4 Pages)

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

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

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.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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HPCI Flow Controller Failure to Achieve Rated Flow while in Auto Mode

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	DOCKET NUMBER
										05000
05	23	2013	2013	006	00	7	22	2013	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE

N

10. POWER LEVEL

002%

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)
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- ☐ 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

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
X	BJ	FIC	N430	Yes					

14. SUPPLEMENTAL REPORT EXPECTED

☐ Yes (If yes, complete 15. *EXPECTED SUBMISSION DATE*) ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

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

On May 23, 2013, at 1050 hours, during plant start up from Refueling Outage (RFO-19) with the reactor at 2% core thermal power, reactor pressure at ~ 525 psig, and the mode switch in the Startup / Hot Standby position, Pilgrim Nuclear Power Station (PNPS) declared the High Pressure Coolant Injection (HPCI) system inoperable due to failure of the HPCI flow indicating controller (FIC-2340-1) to maintain system discharge flow rate above 4250 gpm while in the automatic mode from the Main Control Room during planned post maintenance testing. Limiting Condition for Operation (LCO) actions for Technical Specification (TS) 3.5.C.2 were entered.

The cause of the event was determined to be FIC-2340-1 out of calibration due to degradation of the automatic (null) control/output circuit. Corrective action was taken to troubleshoot, recalibrate and adjust the flow controller. Post work testing verified HPCI flow controller operability.

This event was not risk significant and had no adverse impact on the health and/ or safety of the public.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
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Pilgrim Nuclear Power Station	05000293	2013 –	006	– 00	2 OF 4

NARRATIVE

BACKGROUND:

The Pilgrim Station Core Standby Cooling systems (CSCS) consist of the High Pressure Coolant Injection (HPCI) system, Automatic Depressurization system (ADS), Residual Heat Removal (RHR) System Low Pressure Injection (LPCI) mode, and Core Spray (CS) system. The HPCI system is designed to pump water into the reactor vessel for high pressure core cooling. Although not part of the CSCS, the reactor core isolation cooling (RCIC) system is also designed to pump water into the reactor vessel for high pressure core cooling,

The HPCI System flow indicating controller (FIC-2340-1) installed in the Main Control Room functions to maintain a process flow at a desired set-point. The controller provides for both manual and automatic process control and has an internal set-point control circuit. The controller compares a process variable (HPCI flow from FT-2358) with a control set-point (normally set at 4250 gpm).

Engineering Change EC12967 was issued to replace obsolete GMAC HPCI flow controllers with NUS Instrument Corporation Model PID901-540 flow controllers. These NUS flow controllers were reverse engineered and intended to be equivalent replacements. On 2/24/13 the NUS HPCI System flow controller was installed in the Main Control Room and successfully tested to verify operability.

On May 23, 2013, the plant was starting up from a Refueling Outage (RFO-19). In accordance with Technical Specification (TS) 3.5.C.1, the HPCI System is required to be tested at a reactor pressure of 150 psig to verify system operability. Procedure 8.5.4.3 provides test criteria for system operability and ensures that the system automatically starts and can control flow at or above 4250 gpm. The HPCI system was operated on May 23, 2013 at 0034 hours and met test criteria. Subsequent HPCI system runs were planned to address post maintenance test requirements.

EVENT DESCRIPTION:

On May 23, 2013, at 1050 hours, during plant start-up from RFO-19 with the reactor at 2% core thermal power, reactor pressure at ~ 525 psig, and the mode switch in the Startup/ Hot Standby position, PNPS declared the HPCI system inoperable due to failure of the HPCI flow indicating controller to maintain system discharge flow rate above 4250 gpm while in the automatic mode from the Main Control Room during planned post maintenance testing. Limiting Condition for Operation (LCO) actions for Technical Specification 3.5.C.2 were entered.

CAUSE OF THE EVENT:

The apparent cause evaluation identified that the direct cause of the HPCI system failure was flow controller FIC-2340-1 out of calibration by 550 gpm due to degradation of the flow controller automatic (null) control/output circuit.

The apparent cause evaluation was based on removal of the flow controller, bench testing, and implementing a detailed troubleshooting plan.

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EXTENT OF CONDITION:

PID901-540 flow controllers are installed in the HPCI system, RCIC system, and Control Rod Drive (CRD) system. Based on system testing, the condition was only identified in the HPCI flow controller located in the Main Control Room.

CORRECTIVE ACTIONS:

Corrective actions completed included troubleshooting, bench testing, and successful recalibration and adjustment of the HPCI flow controller. Post calibration testing confirmed stable operation of the flow controller during HPCI System Operability test runs.

Corrective actions planned include:

- Replace the installed Main Control Room HPCI flow controller.
- Send the replaced flow controller to the vendor\manufacturer for evaluation.
- Revise flow controller calibration procedures as necessary to address adequate guidance/steps to check for the degradation that caused this event.
- After vendor evaluation, incorporate appropriate revisions into applicable procedures and document actions in the Corrective Action Program (CAP).

These corrective actions will be tracked in the Corrective Action Program via CR-PNP-2012-4286.

ASSESSMENT OF SAFETY CONSEQUENCES:

The event occurred during power ascension from RFO-19. Core Thermal Power was at approximately 2% and reactor pressure was approximately 525 psig.

CSCS systems include HPCI, ADS, CS, and RHR - LPCI mode. Although not part of the CSCS systems, the RCIC system is capable of providing water to the reactor vessel for high pressure core cooling, similar to the HPCI system.

During the time period that HPCI flow controller was out of service, the ADS, CS, RHR, and RCIC systems were either operable or available. These systems provided capability to supply makeup water to the vessel and ensured adequate core cooling while the HPCI system was not operable. During the event, the HPCI system automatically started and controlled flow at slightly less than 4250 gpm. HPCI system was restored to operable status and there was no long term safety significance associated with the event.

The bounding case of risk assessment was failure of the HPCI pump to operate. This would result in an increase in core damage frequency (CDF) of 3.66E-6/reactor year. The exposure time is estimated from when the last successful run of the HPCI Pump was performed on 5/23/13 at 0034 hrs until the HPCI System and flow controller was tested satisfactorily on 5/24/13 at 0230 hours. This results in approximately 26 hours of exposure time and the incremental core damage probability (ICDP) is 5.43E-9, which is non-risk significant.

REPORTABILITY

This event was reported to the USNRC via Event Report #49064 on 5/23/2013 pursuant to 10 CFR 50.72(b)(3)(v)(B) and (D) - Any event or condition that at the time of discovery could have prevented fulfillment of the safety function of structures or systems that are needed to: Remove residual heat and Mitigate the consequences of an accident.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
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PREVIOUS OCCURRENCES:

A review of Pilgrim Station License Event Reports (LERs) issued since year 2000 was performed. The focus of the review was LERs that involved loss of HPCI system function or loss of system function due to flow controller malfunction. The following LERs were reviewed:

LER 2000-002 - HPCI System Inoperable Due to Power Inverter Feed to Flow Controller Circuitry

LER-2004-002 - HPCI System Inoperable Due to Fuse Failure in Gland Seal Condenser Circuit.

LER 2004-004 - RCIC System Inoperable Due to Flow Controller Oxidation

LER 2005-001 - HPCI System Inoperable Due to Fuse Failure in Motor Operated Valve Control Circuit

LER 2008-004 - HPCI System Inoperable Due to Undervoltage Relay Failure in Valve Power Supply Circuit

LER 2011-006 - HPCI System Inoperable Due to Governor Control Valve Mechanical Binding

These LER events do not identify any similar failure mechanisms to that described in this LER.

In March 2012, Pilgrim Station identified defects in NUS Model PID901-540 flow controllers that were purchased to replace HPCI and RCIC System flow controllers (Condition Report CR-PNP-2012-1406). A manufacturer report was generated to document the 10 CFR Part 21 Evaluation (No. 21-12-09). The issue specifically addressed relates to flow controller setpoint thumbwheel manufacturing assembly defects. Pilgrim sent the all flow controllers back to the manufacturer for reconditioning.

The condition addressed in this LER event report differs from the manufacturing defects evaluated in the vendor's Part 21 evaluation.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for Components and Systems referenced in this report are as follows:

COMPONENTS	CODES
Flow Indicating Controller	FIC
SYSTEMS	
High Pressure Coolant Injection (HPCI)	BJ

REFERENCES:

Condition Report CR-PNP-2013-4286 and the associated Apparent Cause Evaluation Report; HPCI Flow Controller Failure to Achieve Rated Flow While in Auto.

Condition Report CR-PNP-2012-1406, NUS Model 901-540 Flow controller defects.