Entergy

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

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

July 08, 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 2011-003-00, Reactor Scram on Intermediate Range Monitor High-High Flux

LETTER NUMBER: 2.11.043

Dear Sir or Madam:

The enclosed Licensee Event Report (LER) 2011-003-00, "Reactor Scram on Intermediate Range Monitor High-High Flux" is 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,

Steve Bethay for RGS

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

RMB/rmb

Attachment: Licensee Event Report 2011-003-00, Reactor Scram on Intermediate Range Monitor High-High Flux (7 Pages)

IE22

cc:

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USNRC Senior Resident Inspector
Pilgrim Nuclear Power Station

**Attachment 1**
Letter Number 2.11.043

Licensee Event Report 2011-003-00,
Reactor Scram on Intermediate Range Monitor High-High Flux

(7 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 |
|---|---|---|
| **Pilgrim Nuclear Power Station** | **05000293** | **1 OF 7** |

**4. TITLE**
**Reactor Scram on Intermediate Range Monitor High - High Flux**

| 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** |
| **5** | **10** | **2011** | **2011** | **003** | **00** | **7** | **08** | **2011** | FACILITY NAME: **N/A** | DOCKET NUMBER: **05000** |

**9. OPERATING MODE:** N

**10. POWER LEVEL:** 1.7%

**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 |
|---|---|---|---|---|---|---|---|---|---|
| **N/A** | **N/A** | **N/A** | **N/A** | **No** | | | | | |

| 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 May 10, 2011, during STARTUP Mode with reactor thermal power at approximately 1.7 percent, Pilgrim Nuclear Power Station (PNPS) experienced an automatic reactor scram while raising reactor temperature and pressure. The reactor scrammed on intermediate range monitor (IRM) hi-hi flux on both reactor protection system channels. Prior to the scram, operators took the reactor critical, reached the point of adding heat, and established a heatup rate. During the heatup, operators observed a high heatup rate and in response, the shift manager directed operators to insert control rods to reduce the heatup rate. The number of rods or number of notches to insert was not specified. Power began to lower as expected; however, operators did not recognize that inserting control rods to reduce heatup rate with rising moderator temperature caused the reactor to become subcritical. After achieving a temperature change from the power reduction, operators withdrew the same control rods before evaluating the core condition. The resultant reactor response was a faster power ascension rate than expected, which led to an automatic intermediate-range high-flux reactor trip. All systems operated as expected, in accordance with design.

The root cause of this event was determined to be the failure to adhere to established standards and expectations due to a lack of consistent supervisory and management enforcement. Investigation into the event revealed several examples of inconsistent enforcement of administrative procedure requirements and management expectations for command and control, roles and responsibilities, reactivity manipulations, clear communications, proper briefings, and proper turnovers.

Numerous corrective actions have been taken or are planned to reinforce standards and retrain crew members on the fundamentals of criticality and heating range operation.

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

NARRATIVE

**EVENT DESCRIPTION:**

On May 10, 2011, following refueling outage #18 (RFO18), the reactor mode switch was taken to startup at 06:29. Control rod withdrawal for unit startup commenced at 06:41. The control room crew consisted of the following personnel:

- Shift Manager (SM) – management oversight
- Reactivity SRO/Control Room Supervisor (CRS) – command & control
- Assistant Control Room Supervisor (ACRS)
- At-The-Controls (ATC) Operator
- Licensed Operator (Verifier) – control rod verifier
- Reactor Engineer (RE)
- Reactor Engineer in Training
- Assistant Operations Manager – Shift (AOM-Shift) – broad senior management oversight

Additional licensed operators were present in the control room conducting various startup related activities.

At 12:14, the reactor was made critical when control rod 38-19 was moved to position 12. Power ascended to the point of adding heat (POAH) which was achieved at 12:27. Once POAH was achieved, the ATC Operator inserted rod 38-19 to position 10 to obtain IRM overlap correlation data. Following the data collection, the ATC operator withdrew rod 38-19 back to position 12.

The Reactivity SRO/CRS and the ATC Operator continued with plant start-up and were relieved by other qualified licensed operators. The crew withdrew control rods to establish a moderator heatup rate. The ATC Operator withdrew four additional control rods from positions 8 to 12 without incident (8 total notches in 4 minutes).

The ATC Operator continued in the rod pull sequence and encountered difficulty withdrawing control rod 30-11 from position 8 to 12. While attempting to withdraw 30-11, the rod inadvertently settled at position 06. The rod was subsequently moved to position 12.

Following withdrawal of the five control rods (ten control rod notches), the ATC Operator observed the process computer to be displaying a high short-term (5 minute) moderator heatup rate reading of 18°F that corresponded to an approximate 216°F per hour heatup rate (the actual hourly heat-up rate was 50°F/hr). The heatup rate concern was discussed among the SM, Reactivity SRO/CRS, ATC Operator, Verifier and AOM-Shift. After the discussion, the SM directed the at-the-controls crew to insert control rods to reduce the heatup rate. This direction did not include specific guidance or limitations regarding the number of control rod notches to insert.

The RE and RE-in-training were stationed approximately 25 feet from the ATC operator at their computer terminals performing procedurally required calculations related to the startup. The REs had been occupied with these tasks from the time criticality had been achieved and had not been consulted on the plan to insert control rods.

The ATC Operator proceeded to re-insert control rods that were withdrawn to establish the heatup: 30-11, 22-43, 14-19, 38-35 and 14-35 from positions 12 to 8 (10 notches total). At the end of the rod insertion evolution, the SM directed the Reactivity SRO/CRS and the ATC Operator to keep reactor power on IRM range 7.

Reactor power lowered as a result of the control rod insertions, requiring the ATC Operator to range the IRMs down twice. The crew did not recognize that reactor had become subcritical. Moderator temperature was 40°F higher than initial criticality. The higher temperature slightly increased control rod worth.

After approximately a four (4) minute wait from the time the control rods had been inserted, the ATC Operator observed the process computer displaying 0°F per hour heatup rate. At this time, the SRO who had previously been relieved, returned and reassumed his role as Reactivity SRO/CRS. The Reactivity SRO/CRS and the ATC Operator decided to withdraw control rods to establish the desired heatup rate. The same control rods 14-35, 38-35, and 14-19 were withdrawn from positions 8-12 resulting in a rising IRM count rate that was observed by the operators. The change in reactor status was not recognized by the crew. Direct Source Range Monitor (SRM) period indication was not available as the SRMs were previously fully withdrawn in accordance with procedural requirements. Post evaluation showed the reactor period to be 70 seconds.

The ATC Operator continued rod withdrawal with control rod 22-43 from position 8 to 10. The ATC Operator and the Verifier ranged the IRMs up as reactor power increased. The ATC Operator then withdrew control rod 22-43 from position 10 to 12. The operators did not recognize the rate change in IRM power. Post evaluation showed reactor period to be 40 seconds.

In an effort to return all the control rods previously inserted back to their original positions, the ATC Operator selected and inserted control rod 30-11 from position 8 to 10. IRM readings rose at an accelerated rate and an IRM Hi-Hi flux condition was experienced on both RPS channels. Post evaluation showed reactor period to be approximately 20 seconds and APRM power at approximately 1.7% power.

IRM System Status

An immediate verification of the operability and functionality of the IRMs was conducted. The source of the scram signal was from the IRM channel "F" and "G". A review of the plant computer (EPIC) traces for the event determined that the setpoints for RPS actuation had been reached for these channels and that all IRM channels had responded to a neutron flux increase due to operator withdrawal of control rods. All IRM channels had functioned as designed and all setpoints were within procedural tolerance at the time of the event. The IRM system response was determined to be appropriate.

Core and Control Blade Status

Reactor fuel loading and control blade history were reexamined. Configuration and design core conditions were validated as correct.

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

## ROOT CAUSE OF EVENT:

The root cause of this event was determined to be the failure to adhere to established standards and expectations due to a lack of consistent supervisory and management enforcement.

Investigation into the event revealed several examples of inconsistent enforcement of administrative procedure requirements and management expectations for command and control, roles and responsibilities, reactivity manipulations, clear communications, proper briefings, and proper turnovers.

Enforcement of High Standards

For the root cause, there were precursors immediately prior to the event that if acted upon in a timelier or effective manner would have prevented its occurrence. These precursors included a number of missed opportunities just prior to the reactor scram that should have prompted action to reinforce and utilize established procedural standards including, but not limited to, actions for a mispositioned control rod, responding to a high heat up rate, incorporating reactor engineering personnel into reactivity discussions, and effective turnover of reactivity responsibilities.

Other precursors were identified several months prior to the event from site and fleet assessments and observations. These included INPO SOER 10-2 site and fleet 'deep dive' assessments of operator performance and the site observation program. These assessments identified examples of operator standards and expectations shortfalls with regards to procedure noncompliance and communications, as well as supervisors and managers not correcting behaviors. The actions were focused on correcting individual behaviors and failed to ensure that broad departmental actions were taken.

Finally, existing standards contained in "reference use" procedures for the conduct of operations, including those for reactivity manipulations, contain sufficient guidance for individual responsibilities, that when properly implemented, the success of crew activities is assured. Assessment of this event identified shortfalls in the execution of responsibilities for oversight, supervisory direction and orders, engagement of reactor engineering expertise, providing sound concurrent verification, communications, proper turnover, and stopping and resetting via control room briefing.

## CONTRIBUTING CAUSES OF EVENT:

Three contributing causes were identified. Two contributors to this event were related to distinct aspects of operator fundamentals and the other was related to latent procedure weaknesses.

Contributing causes to this event were that operators became overly focused on a single indication and parameter versus reviewing diverse indications, did not display proficiency during the reactor startup in the control of plant parameters and having the correct picture of a successful startup. Additionally, operating procedures lacked detailed guidance for operations from criticality through the IRM heat up range.

Parameter Monitoring

Operators became overly focused on a single indication and parameter versus reviewing diverse indications. In particular, operators did not understand shortfalls associated with the use of the indicator and became overly focused on the short term reactor heat up rate while not integrating input from other heat up rate indications. The undue focus on addressing the perceived excessively high heat up rate led to a lack of focus on other key start up parameters. This weakness led to inserting more notches than necessary to control heat up which ultimately led to the reactor becoming subcritical. Subsequently, using the same indication to re-establish the heat up rate led to withdrawing too many notches too quickly.

Additionally, the operators used the 'five minute average' in a manner that was not intended. The computer point provides a gross trend of moderator temperature by providing the delta temperature between current moderator temperature and that recorded five minutes earlier and not the plant procedure governing heat up and cool down.

Operator Proficiency

The operators did not display proficiency during the reactor startup in the control of plant parameters and having the correct picture of a successful startup.

Specifically, for those operators that did attend pre-startup Just-in-Time training (JITT), the training did not require the operators to operate in the heating range, to use varied methods of determining heat up rate, or understand the limitations of the short term heat up rate instrumentation. Some operators involved in the event did not attend JITT.

Procedure Guidance Not Optimum

Operating procedures lacked detailed guidance for operations from criticality through the IRM heat up range.

After achieving criticality, the guidance for heatup is very limited and instructs the operator to "maintain a heat-up rate not to exceed 100 degrees F / hr when averaged over a one hour period." The step does not specify the instrument(s) to be used to establish the heat-up rate or provide any cautions or limitations required for the use of instrumentation. No direction is provided for addressing the potential for reactor sub criticality or for assessing core conditions prior to reinsertion of positive reactivity following control rod insertion. Additional steps or precautions could have been an effective barrier for this event.

## EXTENT OF CONDITION:

Ineffective reinforcement of standards and expectations has the potential to affect performance in other departments. The Site has initiated actions to reinforce and follow up coaching effectiveness and quality in this regard. A corrective action has been initiated to track implementation of this initiative. Additionally, Site Senior Leadership has an action to reinforce expectations regarding procedure use and adherence and procedure fundamentals.

**FAILED COMPONENT IDENTIFICATION:**

Not applicable.

**CORRECTIVE ACTIONS:**

Immediate corrective actions taken as a result of the event including those to reduce the possibility of similar events occurring in the future include:

- Retrained crew members on criticality and heating range operation.
- Disqualified involved individuals pending investigation, accountability assessment and remediation.
- Established interim guidance requiring Operations Management, Reactor Engineering, and Crew review following any negative reactivity addition in STARTUP prior to adding positive reactivity (Standing Order 11-04) prior to 5/11/2011 startup.
- Revised & implemented startup JITT to include lessons learned managing heatup.
- Established immediate Operations Management oversight for 100% of the time in Startup Mode.
- Implemented station senior management oversight program for duration of RFO18 startup
- Implemented Fleet Significant Event Response Team (SERT)
- Coached Reactor Engineers on standards & expectations regarding advocacy & intrusiveness during reactivity manipulations
- Corrected deficiency associated with control rod 30-11.

Additional significant corrective actions planned as a result of the event include:

- Implement 200 hours of plant operations department management (including cross crew) control room observations using Entergy Fleet Procedure EN-OP-117 and identify gaps to established standards.
- Perform assessments of operations performance and observations from coaching and other oversight input. Review results of observations and corrective actions taken monthly with Site Directors.
- Revise JITT to make heating range a required element of the module and require successful completion of evaluation.
- Revise startup JITT to require inclusion of all individuals having a significant role in the evolution (including reliefs) and support personnel.

These and additional corrective actions are being tracked in the PNPS Corrective Action Program.

**ASSESSMENT OF SAFETY CONSEQUENCES:**

An assessment of safety consequences and implications of the event found ineffective adherence to standards and expectations and the inability to apply fundamental knowledge during reactivity manipulations represent a serious challenge to safe plant operation. This event however, did not challenge safety limits or fission product barriers. Core power was maintained within design limits and the control rod pattern remained within the bounds of the Banked Position Withdraw Sequence (BPWS). There were also no radiological or industrial safety challenges associated with this event and posed no threat to public health and safety

## SIMILAR EVENTS:

Over the past five (5) years, Pilgrim Station has not experienced a similar adverse consequence during reactor startup activities.

A detailed search of the Entergy Fleet Paperless Condition Reporting System (PCRS) and the INPO Operating Experience (OE) database was conducted for operating experience concerning reactor scrams during startup activities to ascertain whether there has been any commonality to the IRM Hi-Hi Flux scram experienced at Pilgrim Station.

The search found numerous examples of industry operating experience which were applicable to this event. While the causes of this event indicate shortfalls in execution consistent with established standards instead of process weaknesses, site management considers it prudent to perform additional reviews of INPO SOER 10-2 and by extension INPO SOERs, 07-01 and 96-01. The review will identify any additional corrective actions that may be required. A corrective action has been assigned within the root cause analysis to conduct this review.

## REFERENCES:

Condition Report associated with Root Cause Analysis; Reactor Scram on IRM Hi-Hi Flux (This condition report contains reference to various station procedures and all condition reports written during the event)