



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

September 1, 2011

EA-11-174

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

SUBJECT: PILGRIM NUCLEAR POWER STATION - NRC SPECIAL INSPECTION
REPORT 05000293/2011012; PRELIMINARY WHITE FINDING

Dear Mr. Smith:

On July 20, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed a Special Inspection at your Pilgrim Nuclear Power Station (PNPS). The inspection was conducted in response to the May 10, 2011, reactor scram event that occurred due to an unrecognized subcriticality and subsequent unrecognized return to criticality. The NRC's initial evaluation of this event satisfied the criteria in NRC Inspection Manual Chapter (IMC) 0309, "Reactive Inspection Decision Basis for Reactors," for conducting a Special Inspection. The Special Inspection Team (SIT) Charter (Attachment 2 of the enclosed report) provides the basis and additional details concerning the scope of the inspection. The enclosed inspection report documents the inspection results, which were discussed at the exit meeting on July 20, 2011, with you and other members of your staff.

The inspection team examined activities conducted under your license as they relate to safety and compliance with Commission rules and regulations and with the conditions of your license. The inspection team reviewed selected procedures and records, observed activities, and interviewed personnel. In particular, the inspection team reviewed event evaluations, causal investigations, relevant performance history, and extent of condition to assess the significance and potential consequences of issues related to the May 10 event.

The inspection team concluded that the plant operated within acceptable power limits, and no equipment malfunctioned during the power transient and subsequent reactor scram. Nonetheless, the inspection team identified several issues related to human performance and compliance with conduct of operations and reactivity control standards and procedures that contributed to the event. The enclosed chronology (Attachment 3 of the enclosed report) provides additional details regarding the sequence of events.

This report documents one finding that, using the reactor safety Significance Determination Process (SDP), has preliminarily been determined to be White, or of low to moderate safety significance. The finding involves the failure of Pilgrim personnel to implement conduct of operations and reactivity control standards and procedures during a reactor startup, which contributed to an unrecognized subcriticality followed by an unrecognized return to criticality and subsequent reactor scram.

This finding was assessed using NRC IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," because probabilistic risk assessment tools were not well suited to evaluate the multiple human performance errors associated with this issue. Preliminarily, the NRC has determined this finding to be of low to moderate safety significance based on a qualitative assessment. There was no significant impact on the plant following the transient because the event itself did not result in power exceeding license limits or fuel damage. Additionally, interim corrective actions were taken, which included removing the Pilgrim control room personnel involved in the event from operational duties pending remediation, providing additional training for operators not involved with the event, and providing increased management oversight presence in the Pilgrim control room while long term corrective actions were developed.

The finding involved one apparent violation (AV) of NRC requirements regarding Technical Specification 5.4, "Procedures," that is being considered for escalated enforcement action in accordance with the NRC's Enforcement Policy, which can be found on NRC's website at <http://www.nrc.gov/reading-rom/doc-collections/enforcement/>.

In accordance with NRC IMC 0609, we will complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of the date of this letter. The SDP encourages an open dialogue between the NRC staff and the licensee; however, the dialogue should not impact the timeliness of the staff's final determination. Before we make a final decision on this matter, we are providing you with an opportunity to (1) attend a Regulatory Conference where you can present to the NRC your perspective on the facts and assumptions the NRC used to arrive at the finding and assess its significance, or (2) submit your position on the finding to the NRC in writing. If you request a Regulatory Conference, it should be held within 30 days of your response to this letter, and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. If you decide to submit only a written response, such submittal should be sent to the NRC within 30 days of your receipt of this letter. If you decline to request a Regulatory Conference or submit a written response, you relinquish your right to appeal the final SDP determination, in that by not doing either, you fail to meet the appeal requirements stated in the Prerequisite and Limitation Sections of Attachment 2 of IMC 0609.

Please contact Mr. Donald E. Jackson by telephone at (610) 337-5306 within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

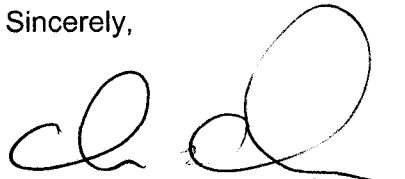
R. Smith

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Because the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. Please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room and from the Publicly Available Records (PARS) component of NRC's document system, 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,



Christopher G. Miller, Director
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000293/2011012
w/Attachments: Supplemental Information (Attachment 1)
Special Inspection Team Charter (Attachment 2)
Detailed Sequence of Events (Attachment 3)
Appendix M Table 4.1 (Attachment 4)

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Sincerely,

/RA/

Christopher G. Miller, Director
Division of Reactor Safety

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.: 50-293

License No.: DPR-35

Report No.: 05000293/2011012

Licensee: Entergy Nuclear Operations, Inc

Facility: Pilgrim Nuclear Power Station (PNPS)

Location: 600 Rocky Hill Road
Plymouth, MA 02360

Dates: May 16 through July 20, 2011

Team Leader: R. McKinley, Senior Emergency Response Coordinator
Division of Reactor Safety

Team: B. Haagensen, Resident Inspector, Division of Reactor Projects
D. Molteni, Operations Engineer, Division of Reactor Safety

Approved By: Donald E. Jackson, Chief
Operations Branch
Division of Reactor Safety

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SUMMARY OF FINDINGS

IR 05000293/2011012; 05/16/2011 - 07/20/2011; Pilgrim Nuclear Power Station (PNPS); Inspection Procedure 93812, Special Inspection.

A three-person NRC team, comprised of two regional inspectors and one resident inspector, conducted this Special Inspection. One finding with potential for greater than Green safety significance was identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspect was determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

NRC Identified and Self Revealing Findings

Cornerstone: Initiating Events

- **Preliminary White:** A self-revealing finding was identified involving the failure of Pilgrim personnel to implement conduct of operations and reactivity control standards and procedures during a reactor startup, which contributed to an unrecognized subcriticality followed by an unrecognized return to criticality and subsequent reactor scram.

The significance of the finding has preliminarily been determined to be White, or of low to moderate safety significance. The finding is also associated with one apparent violation of NRC requirements specified by Technical Specification 5.4, "Procedures." There was no significant impact on the plant following the transient because the event itself did not result in power exceeding license limits or fuel damage. Additionally, interim corrective actions were taken, which included removing the Pilgrim control room personnel involved in the event from operational duties pending remediation, providing additional training for operators not involved with the event, and providing increased management oversight presence in the Pilgrim control room while long term corrective actions were developed. Entergy staff entered this issue, including the evaluation of extent of condition, into its corrective action program (CR-PNP-2011-2475) and performed a Root Cause Evaluation (RCE).

The finding is more than minor because it was associated with the Human Performance attribute of the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Specifically, the failure of Pilgrim personnel to effectively implement conduct of operations and reactivity control standards and procedures during a reactor startup caused an unrecognized subcriticality followed by an unrecognized return to criticality and subsequent reactor scram. Because the finding primarily involved multiple human performance errors, probabilistic risk assessment tools were not well suited for evaluating its significance. The inspection team determined that the criteria for using IMC 0609, Appendix M, "Significance Determination Process Using

Qualitative Criteria," were met, and the finding was evaluated using this guidance, as described in Attachment 4 to this report. Based on the qualitative review of this finding, the NRC has preliminarily concluded that the finding was of low to moderate safety significance (preliminary White).

The inspection team determined that multiple factors contributed to this performance deficiency, including: inadequate enforcement of operating standards, failure to follow procedures, and ineffective operator training. The Entergy RCE determined that the primary cause was a failure to adhere to established Entergy standards and expectations due to a lack of consistent supervisory and management enforcement. The inspection team concluded that the finding had a cross-cutting aspect in the Human Performance cross-cutting area, Work Practices component, because Entergy did not adequately enforce human error prevention techniques, such as procedural adherence, holding pre-job briefs, self and peer checking, and proper documentation of activities during a reactor startup, which is a risk significant evolution. Additionally, licensed personnel did not effectively implement the human performance prevention techniques mentioned above, and they proceeded when they encountered uncertainty and unexpected circumstances during the reactor startup [H.4(a)]. (Section 2)

REPORT DETAILS

1. Background and Description of Event

In accordance with the Special Inspection Team (SIT) Charter (Attachment 2), the inspection team conducted a detailed review of the May 10, 2011, reactor scram event at Pilgrim Nuclear Power Station, including a review of the Pilgrim operators' response to the event. The inspection team gathered information from the plant process computer (PPC) alarm printouts and parameter trends, interviewed station personnel, observed on-going control room activities, and reviewed procedures, logs, and various technical documents to develop a detailed timeline of the event (Attachment 3).

On May 10, 2011, following a refueling outage, the reactor mode switch was taken to startup at 0626, and control rod withdrawal commenced at 0641. The control room crew consisted of the following personnel (additional licensed operators were present in the control room conducting various startup related activities):

- Assistant Operations Manager (AOM-Shift) – Senior Line Management oversight
- Shift Manager (SM) – management oversight
- Reactivity Senior Reactor Operator (SRO)/Control Room Supervisor (CRS) – command and control
- Assistant Control Room Supervisor (ACRS)
- Reactor Operator At-The-Controls (RO-ATC)
- Reactor Operator (Verifier) – ATC verifier
- Reactor Engineer (RE)
- RE in Training

At 1212, the reactor was made critical when control rod 38-19 was moved to position 12. Power continued to rise to the point of adding heat (POAH), and the POAH was achieved at 1227. Once the POAH was achieved, the RO-ATC operator inserted rod 38-19 to position 10 to obtain Intermediate Range Monitor (IRM) overlap correlation data. Following the data collection, the RO-ATC operator withdrew rod 38-19 back to position 12.

At approximately 1231, the Reactivity SRO/CRS and the RO-ATC operator were relieved by other licensed operators who continued with plant startup. The crew withdrew control rods to establish a moderator heat-up rate. The RO-ATC operator withdrew control rods 14-35, 38-35, 14-19 and 22-43 from position 08 to 12 without incident.

The RO-ATC operator continued with the rod withdrawal sequence and tried to withdraw control rod 30-11 from position 08 to 12, but the control rod would not move using normal notch withdraw commands. The RO-ATC then attempted to withdraw control rod 30-11 using a "double-clutch" maneuver in accordance with procedures; however, the control rod inadvertently inserted and settled at position 06. As stated during interviews with the NRC inspectors, the RO-ATC operator, the ATC verifier, and the Reactivity SRO/CRS all saw the control rod in the incorrect position. However, the operators did not enter and follow Pilgrim Nuclear Power Station (PNPS) Procedure 2.4.11, "Control Rod Positioning Malfunctions" as required. This procedure required the operators to assess the amount of the mispositioning to determine the appropriate course of remedial

action before proceeding, and it also required the issue to be documented in a condition report. The operators did not perform an assessment, and they moved the control rod back to position 08 and ultimately to position 12, which was the correct final position in accordance with reactor engineering maneuvering instructions. During interviews with the NRC inspectors, the three operators each indicated that there was confusion in their mind regarding whether or not the control rod met the definition of a mispositioned control rod because the control rod was only out of position by one notch from the initial position, but none of the operators referred to the procedure, and there was no discussion or challenge regarding the proper course of action among the operators. The condition was not logged, and a condition report was not generated until the issue was identified by NRC inspectors. In addition, the problem of the mispositioned control rod was not discovered by the licensee during the post trip review.

Following withdrawal of the five control rods (ten control rod notches), the RO-ATC observed the process computer displaying a high short-term (five minute average) moderator heat-up rate reading of 18°F per 5 minutes that he mistakenly believed corresponded to an hourly heat-up rate of 216°F/hr (the actual hourly heat-up rate was 50°F/hr). The heat-up rate concern was discussed among the SM, Reactivity SRO/CRS, RO-ATC operator, Verifier and AOM-Shift. After the discussion, the SM directed the crew at the controls to insert control rods to reduce the heat-up rate. This direction did not include specific guidance or limitations regarding the number of control rod notches to insert. At this point, the AOM-Shift and SM left the front panels area of the control room.

The RE and RE-in-training were working at their computer terminals in the control room 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 to reduce the heat-up rate. The RE-in-training overheard the operator conversation about inserting control rods. He informed the RE, who in turn, questioned the SM about the decision to insert rods. The SM responded that the actions were necessary to control heat-up rate. No further discussion occurred between the SM and the RE regarding the number of control rods/notches to be used to control the heat-up rate or if there was a need to modify the reactor maneuvering plan. During interviews with the NRC inspectors, the SM and the AOM-Shift stated that they both discussed that there was a need to be careful to avoid taking the reactor subcritical and that the action of inserting control rods had the potential to cause the reactor to become subcritical. However, this important information was never communicated to any of the operators at the controls, including at the time when the SM directed the at-the-controls crew to insert control rods to reduce the heat-up rate.

As a result of the previous control rod withdrawal, moderator temperature was 40°F higher than it was at initial criticality resulting in slightly increased control rod worth. The crew did not factor this increased control rod worth into their decision regarding the number of control rod notches to insert.

Over the next three minutes, the RO-ATC operator proceeded to re-insert the following control rods from positions 12 to 8 (10 notches total) that had been previously withdrawn

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to establish the heat-up: 30-11, 22-43, 14-19, 38-35 and 14-35. At the end of the rod insertion evolution, the SM directed the Reactivity SRO/CRS and the RO-ATC operator to keep reactor power on IRM range 7. This communication was not acknowledged by the RO-ATC operator. During interviews with the NRC inspectors, none of the operators recalled receiving such instructions. The SM then left the control room to take a break. The AOM-Shift left the controls area to get lunch in the control room kitchen.

As a result of the control rod insertions, reactor power lowered, thus requiring the RO-ATC operator to range the IRMs down to range 7 and then to range 6. The reactor had become subcritical, but the crew did not recognize the change in reactor status.

Approximately four minutes after the control rods were inserted to reduce the heat-up rate, the RO-ATC operator observed the process computer displaying a 0°F/hr heat-up rate. At this time, the SRO who had previously been relieved, returned and re-assumed his role as Reactivity SRO/CRS. The Reactivity SRO/CRS and the RO-ATC operator decided to once again withdraw control rods to re-establish the desired heat-up rate. Three of 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. However, the crew did not recognize that the reactor status had changed from subcritical to critical.

At this point, the AOM-Shift returned to the reactor panel area. The RO-ATC operator continued rod withdrawal with control rod 22-43 from position 08 to 10. The RO-ATC operator and the Verifier ranged the IRMs up as reactor power increased. The RO-ATC operator then withdrew control rod 22-43 from position 10 to 12. The operators did not recognize the increasing rate of change in IRM power.

Finally, the RO-ATC operator selected and withdrew control rod 30-11 from position 8 to 10. At 1318, IRM readings rose sharply and an IRM Hi-Hi flux condition was experienced on both Reactor Protection System (RPS) channels resulting in an automatic reactor scram at approximately 1.7% reactor power.

2. Operator Human Performance

a. Inspection Scope

The inspection team interviewed the Pilgrim control room personnel that responded to the May 10, 2011, event including the SM, AOM-Shift, CRS, ACRS, RO-ATC, RO verifier, and the REs to determine whether these personnel performed their duties in accordance with plant procedures and training. The inspection team also reviewed narrative logs, sequence of events and alarm printouts, condition reports, PPC trend data, procedures implemented by the crew, and procedures regarding the conduct of operations.

b. Findings/Observations

Failure to Implement Procedures during Reactor Startup

Introduction: A self-revealing finding was identified involving the failure of Pilgrim personnel to implement conduct of operations and reactivity control standards and procedures during a reactor startup, which contributed to an unrecognized subcriticality followed by an unrecognized return to criticality and subsequent reactor scram. The significance of the finding has preliminarily been determined to be White, or of low to moderate safety significance. The finding is also associated with one apparent violation of NRC requirements specified by Technical Specification 5.4, "Procedures."

Description: On May 10, 2011, following a refueling outage, operators were in the process of conducting a reactor startup. During the course of the startup, multiple licensed operators failed to implement written procedures as described below:

- Entergy procedure EN-OP-115, "Conduct of Operations," Revision 10, Section 4.0, states that the SM is to "provide oversight of activities supporting complex and infrequently performed plant evolutions such as plant heat-up [and] startup." Additionally, the SM is responsible for ensuring "conservative actions are taken during unusual conditions ... when dealing with reactivity control." However, the SM did not oversee the activities in progress during reactor heat-up and left the control room when the heat-up rate was being adjusted with control rod insertion. The SM did not ensure the actions taken to reestablish or adjust the reactor heat-up rate were conservative nor did he reinforce those actions with the operating crew.
- Entergy procedure EN-OP-115, "Conduct of Operations," Revision 10, Section 4.0, states that the CRS is required to "Ensure Pre-Evolution Briefings are held [and] plant operations are conducted in compliance with administrative and regulatory requirements." PNPS procedure 1.3.34, "Operations Administrative Policies and Procedures," Revision 117, Section 6.10.1.1 states, "All complex or infrequently performed activities warrant a pre-evolution briefing." Section 6.10.1.1[8] lists an Infrequently Performed Tests or Evolutions Briefing as one type of pre-evolution briefing, and Section 6.10.1.1[4] states, "Infrequently Performed Tests or Evolutions Briefings for the performance of Procedures classified as "Infrequently Performed Tests or Evolutions" (IPTE) should be performed with Senior Line Manager oversight as specified in EN-OP-116, "Infrequently Performed Tests or Evolutions." Entergy Procedure EN-OP-116, Revision 7, Attachment 9.1 identifies "Reactor Startup" as an IPTE. However, in this case, the licensee conducted a reactor startup without performing an IPTE briefing or any other type of pre-evolution briefing as defined in PNPS procedure 1.3.34. It is noteworthy to point out that an IPTE briefing package was previously prepared, approved, and scheduled; however, the IPTE briefing was never performed as required by the procedures described above. In addition, an IPTE briefing was also not performed for the startup following this event. Finally, the CRSs did not ensure the administrative requirements of the conduct of operations procedures or the regulatory requirement to implement the control rod mispositioning procedure were met. This issue was identified by the NRC inspectors.

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- Entergy procedure EN-OP-115, "Conduct of Operations," Revision 10, Section 5.2, states control room operators are required to "develop and implement a plan that includes contingencies and compensatory measures" and when implementing those plans the "crew ... continuously evaluates the plan for changing conditions" and "Human Performance (HU) tools (... , peer/cross-checking, oversight, questioning attitude, etc.) are utilized ...". In addition, "When the control room team is faced with a time critical decision: Use all available resources...do not proceed in the face of uncertainty..." However, the control room operators failed to develop contingency plans or compensatory measures for adjusting reactor heat-up rate or addressing higher than expected reactor heat-up rates. The crew also failed to develop or implement contingencies for control rods which were difficult to maneuver when they were at low reactor power. Additionally, the use of human performance tools was ineffective in addressing the actions or conditions that led to the unexpected reactor heat-up rate and the mispositioning of control rod 30-11. Specifically, failures in the use of peer checking and questioning the conditions that led to the unexpected reactor heat-up rate directly contributed to the mispositioned control rod and the subsequent reactor scram. Lastly, the control room team did not use all available resources by involving Reactor Engineering staff in its decision-making, and proceeded in the face of uncertainty by failing to consider the consequences of the reactivity changes.
- Entergy procedure EN-OP-115, "Conduct of Operations," Revision 10, Section 5.4 states that reactor operators are expected to perform reactivity manipulations "in a deliberate, carefully controlled manner while the reactor is monitored to ensure the desired result is obtained." However, the reactor operators did not adequately monitor the conditions of the reactor while attempting to establish and adjust the reactor heat-up rate. Although the reactor operators were watching the response of both the IRMs and the computer point displaying a five minute average reactor heat-up, they were moving control rods faster than the plant temperature could respond and therefore taking actions to continue control rod movement before the desired result of their manipulations could be assessed. Additionally, after inserting control rods to adjust the reactor heat-up rate, the operators had sufficient indications that the reactor was significantly subcritical as evidenced by the required ranging down of IRMs, the drop in Source Range Monitor (SRM) count rates, and establishing a negative reactor period. The operator's failure to adequately monitor the status of the reactor led to an unrecognized subcritical condition and subsequent return to criticality resulting in an eventual reactor scram.
- PNPS procedure 1.3.34, "Operations Administrative Policies and Procedures," Revision 117, Section 6.7.5 states, "Any relief occurring during the shift (either short-term or for the remainder of the shift) will be recorded in the CRS log." It further states, "...a verbal discussion of plant status and off-normal conditions must be conducted." However, several people in watch standing positions changed from the start of the shift, but none of those changes were entered into the control room log. In addition, when the ACRS was turning over to the CRS, there was no discussion of the mispositioning of control rod 30-11.

- PNPS Procedure 2.4.11, "Control Rod Positioning Malfunctions," Revision 35, Section 5.4 defines a mispositioned control rod as "a control rod found to be left in a position other than the intended position **OR** a control rod that moves more than one notch beyond its intended position." Attachment 4 Step [3] and Step [4] of the same procedure requires the operators to assess the degree of mispositioning and take the appropriate remedial action depending on the degree of mispositioning. Attachment 4 Step [5] also states, "If the control rod is determined to be mispositioned, then record the event as a condition report." In this case, the RO-ATC attempted to withdraw control rod 30-11 from position 08 to position 10 (intended position), but the rod inadvertently inserted to position 06. Upon recognizing the error, the operators did not enter the procedure when control rod 30-11 was found to be left in a position other than the intended position and which was more than one notch from the intended position. The operators did not assess the amount of the control rod mispositioning in accordance with the procedure, nor was there any discussion about the mispositioning on the crew. Furthermore, the event was not logged, nor was a condition report generated. Instead, the operators did not enter and follow the procedure, and they continued on with the startup in the face of uncertainty. This issue was not detected during the licensee post-trip review. It was identified by the NRC inspectors.
- PNPS Procedure 2.1.1, "Startup from Shutdown," Revision 173, Page 53, Caution 2 states, "In the event the reactor goes subcritical after achieving initial criticality, then return to step [53] and re-perform the steps to restore the Reactor to a critical condition." In addition, PNPS Procedure 2.1.4, "Approach to Critical," Revision 26, Section 5.0 states, "In the event the reactor goes subcritical after achieving initial criticality, then with Reactor Engineering guidance, re-perform Section 7.0 Steps [6] and [7] to restore the Reactor to a critical condition." However, the operators did not recognize that the reactor had become subcritical and did not re-perform the procedural steps mentioned above to restore the reactor to a critical condition in a controlled manner under the guidance of Reactor Engineering. There was sufficient information available to the operators to identify that the reactor had become subcritical. In addition, REs were available in the control room, but they were not consulted by the operators.

Analysis: The inspection team determined that the failure of Pilgrim personnel to implement conduct of operations and reactivity control standards and procedures during a reactor startup was a performance deficiency that was reasonably within Entergy's ability to foresee and prevent. The finding is more than minor because it was associated with the Human Performance attribute of the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Specifically, the failure of Pilgrim personnel to effectively implement conduct of operations and reactivity control standards and procedures during a reactor startup caused an unrecognized subcriticality followed by an unrecognized return to criticality and subsequent reactor scram.

The inspection team determined that multiple factors contributed to this performance deficiency including: inadequate enforcement of operating standards, failure to follow procedures, and ineffective operator training. The Entergy RCE documented that the primary cause was a failure to adhere to established Entergy standards and expectations due to a lack of consistent supervisory and management enforcement. In addition, the Entergy RCE specified a number of condition reports and self assessment reports written in the months preceding this event that demonstrated that the performance deficiency existed over an extended period of time and affected all operating crews. While the performance deficiency manifested itself during this particular low power event, there was the potential for the performance deficiency to result in a more consequential event under different circumstances.

Because the finding primarily involved multiple human performance errors, probabilistic risk assessment tools were not well suited for evaluating its significance. The inspection team determined that the criteria for using IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," were met, and the finding was evaluated using this guidance as described in Attachment 4 to this report. Based on the qualitative review of this finding, the NRC concluded that the finding was preliminarily of low to moderate safety significance (preliminary White). The completed Appendix M table is attached to this report (Attachment 4). There was no significant impact on the plant following the transient because the event itself did not result in power exceeding license limits or fuel damage. Additionally, interim corrective actions were taken, which included removing the Pilgrim control room personnel involved in the event from operational duties pending remediation, providing additional training for operators not involved with the event, and providing increased management oversight presence in the Pilgrim control room while long term corrective actions were developed.

This finding had a cross-cutting aspect in the Human Performance cross-cutting area, Work Practices component, because Entergy management and supervision did not adequately enforce human error prevention techniques, such as procedural adherence, holding pre-job briefs, self and peer checking, and proper documentation of activities during a reactor startup, which is a risk significant evolution. Additionally, licensed personnel did not effectively implement the human performance prevention techniques mentioned above, and they proceeded when they encountered uncertainty and unexpected circumstances during the reactor startup [H.4(a)].

Enforcement: Technical Specification 5.4, "Procedures," states, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix "A" of Regulatory Guide (RG) 1.33, February, 1978. RG 1.33, Appendix "A," requires that typical safety-related activities listed therein be covered by written procedures. Contrary to the above, on May 10, 2011, as reflected in the examples listed in the description section of this finding, the licensee failed to implement safety-related procedures related to RG 1.33, Appendix "A," Paragraph 1, "Administrative Procedures;" Paragraph 2, "General Plant Operating Procedures;" and, Paragraph 4, "Procedures for Startup, Operation, and Shutdown of Safety-Related BWR Systems."

Following a review of the event, the licensee documented the condition in the corrective action program (CR-PNP-2011-2475). There was no significant impact on the plant following the transient because the event itself did not result in power exceeding license limits or fuel damage. Additionally, interim corrective actions were taken, which included removing the Pilgrim control room personnel involved in the event from operational duties pending remediation, providing additional training for operators not involved with the event, and providing increased management oversight presence in the Pilgrim control room while long term corrective actions were developed.

Pending determination of final safety significance, this finding with the associated apparent violation will be tracked as **AV 05000293/2011012-01, Failure to Implement Conduct of Operations and Reactivity Control Procedures during Reactor Startup.**

3. Fitness for Duty

a. Inspection Scope

The inspection team interviewed the control room personnel that were directly involved with the May 10, 2011, reactor scram event as well as management personnel involved with the immediate post event investigation. The inspection team also reviewed Entergy Fitness for Duty (FFD) program requirements contained in the corporate and site procedures.

b. Findings/Observations

No findings were identified.

4. Training

a. Inspection Scope

The inspection team interviewed personnel, reviewed simulator modeling and performance, and reviewed training material related to Just in Time Training (JITT) material for the initial and subsequent startups, remedial training for the operators involved with the event, and training plans for startups and reactivity maneuvers.

b. Findings/Observations

No findings were identified.

The inspection team observed that the JITT training that was provided prior to the initial startup was very limited in scope in that it only covered the approach to criticality up to the POAH. It did not cover the full range of reactor heat-up, and it covered very little Operating Experience. In addition, several operators that were directly involved with this event did not attend the JITT training including the SM, the ACRS who temporarily relieved the CRS prior to the scram, and the RO who was at the controls when the scram occurred.

Enclosure

5. Organizational Response

5.1 Immediate Response

a. Inspection Scope

The inspection team interviewed personnel, reviewed various procedures and records, and observed control room operations to assess immediate response of station personnel to the reactor scram event.

b. Findings/Observations

No findings were identified.

The inspection team observed that Entergy's initial response to the event was not appropriately thorough and was narrowly focused. Immediately following the event, operators were debriefed in an attempt to ascertain the cause of the event. Initially, Entergy personnel focused on a potential IRM malfunction as the potential cause of the event despite the fact that multiple IRM channels accurately tracked reactor power along with operator reactivity inputs. Immediate post event interviews with the crew did not probe human error as a potential cause even though the SM, the AOM-Shift, and the REs had expressed concerns just prior to the scram regarding the insertion of control rods so near the point of criticality. Operators involved with the event were dismissed for the day as the investigation continued to incorrectly focus on equipment malfunction as the most likely cause of the event. Several hours passed before it became clear to site management that human error was the cause of the event. As a result, the operators involved with the event were not thoroughly interviewed to ensure that all of the human performance aspects were fully understood prior to proceeding with the next startup. In addition, the inspection team identified that the post-trip review failed to identify that a control rod had been mispositioned just prior to the scram and that an IPTE briefing had not been conducted for the startup. Consequently, additional human performance issues were not evaluated, and the licensee again failed to perform an IPTE briefing prior to the subsequent startup as required by Entergy procedures.

5.2 Post-Event Root Cause Evaluation and Actions

a. Inspection Scope

The inspection team reviewed Entergy's Root Cause Evaluation (RCE) report for the event to determine whether the causes and associated human performance issues were properly identified. Additionally, the inspection team assessed whether interim and planned long term corrective actions were appropriate to address the cause(s).

b. Findings/Observations

No findings were identified.

The RCE was thorough and appeared to identify the underlying causal factors. The associated proposed corrective actions appeared to adequately address the underlying causal factors. Entergy identified the root cause as a lack of consistent supervisory and management enforcement of administrative procedure requirements and management expectations for command and control, roles and responsibilities, reactivity manipulations, clear communications, proper briefings, and proper turnovers.

The RCE also identified contributing causes including weaknesses in monitoring plant status and parameters as well as weaknesses in operator proficiency with regards to low power operations.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On July 20, 2011, the inspection team discussed the inspection results with Mr. R. Smith, Site Vice President, and members of his staff. The inspection team confirmed that proprietary information reviewed during the inspection period was returned to Entergy.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

R. Smith	Site Vice President
J. Dreyfuss	General Manager Plant Operations
D. Noyes	Manager, Operations
J. Macdonald	Assistant Manager, Operations
R. Probasco	Shift Manager, Operations
J. Couto	Shift Supervisor, Operations
S. Anderson	Shift Supervisor, Operations
T. Tomon	Reactor Operator, Operations
J. Byron	Reactor Operator, Operations
J. Hayhurst	Reactor Operator, Operations
S. Bethay	Director, Nuclear Safety Assurance
J. Lynch	Manager, Licensing
T. White	Manager, Quality Assurance
F. McGinnis	Engineer, Licensing
R. Byrne	Senior Engineer, Licensing
V. Fallacara	Director, Engineering
S. Reininghaus	Manager, Training
J. House	Supervisor, Operations Training
V. Magnatta	Lead Instructor, Operations Training
R. Paranjape	Reactor Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000293/2011012-01	AV	Failure to Implement Conduct of Operations and Reactivity Control Procedures during Reactor Startup (Section 2)
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LIST OF DOCUMENTS REVIEWED

Procedures:

1.3.34, "Operations Administrative policies and Procedures," Revision 119
1.3.37, "Post-Trip Reviews," Revision 27
1.3.63, "Conduct of Event Review Meetings," Revision 25
1.3.109, "Issue Management," Revision 8
2.1.1, "Startup from Shutdown," Revision 173
2.1.4, "Approach to Critical," Revision 26
2.1.7, "Vessel Heat-up and Cool Down," Revision 54
2.4.11, "Control Rod Positioning Malfunctions," Revision 35
2.4.11.1, "CRD System Malfunctions," Revision 21

SUPPLEMENTAL INFORMATION

NOP96A3, "Reactivity Management Peer Panel," Revision 10
EN-FAP-AD-001, "Fleet Administrative Procedure (FAP) Process," Revision 0
EN-FAP-OM-006, "Working Hour Limits for Non-Covered Workers," Revision 2
EN-FAP-OP-008, "Reactivity Management Performance Indicator Program," Revision 0
EN-FAP-OP-011, "Operator Human Performance Indicator Program," Revision 0
EN-HU-102, "Human Performance Tools," Revision 5
EN-HU-103, "Human Performance Error Reviews," Revision 4
EN-NS-102, "Fitness for Duty Program," Revision 9
EN-OM-119, "On-Site Safety Review Committee," Revision 7
EN-OM-123, "Fatigue Management Program," Revision 3
EN-OP-103, "Reactivity Management Program," Revision 5
EN-OP-115, "Conduct of Operations," Revision 10
EN-OP-116, "Infrequently Performed Tests of Evolutions," Revision 7
EN-RE-214, "Conduct of Reactor Engineering," Revision 0
EN-RE-215, "Reactivity Maneuver Plan," Revision 1
EN-RE-219, "Startup sequence Criticality Controls (BWR)," Revision 0

Condition Reports:

CR-PNP-2011-02475 and associated Root Cause Evaluation Report, Revision 1
CR-PNP-2011-02488
CR-PNP-2011-02493
CR-PNP-2011-02504
CR-PNP-2011-02506
CR-PNP-2011-02546
CR-PNP-2011-02568
CR-PNP-2011-02572
CR-PNP-2011-02577
CR-PNP-2011-03598

Self Assessments:

LO-PNPLO-2009-00071, "Focused Assessment on Reactivity Management"
LO-PNPLO-2010-00106, "Snapshot Assessment on Reactivity Management Procedure
Revision Implementation"
LO-PNPLO-2010-00106, "Snapshot Assessment on SOER 07-01 Recommendation 4 Reactivity
Management Operations Training"

Technical Specifications:

3.5.C, "HPCI System"
3.5.D, "RCIC System"
5.4.1, "PROCEDURES"

Training Material:

Instructional Module, Reactor Startup and Criticality (& Main Turbine Overspeed) Just in Time
Training used for 05/10/2011 and 05/11/2011 Startup JITT
Instructional Module, Reactor Startup and Criticality May 2011 Just in Time Training used for
05/18/2011 Startup JITT

SUPPLEMENTAL INFORMATION

Just in Time Training PowerPoint used for 05/18/2011 Startup JITT
Instructor Lesson Plan JITT RFO 18 Hydro 2.1.8.5
Simulator JITT Reactor Shutdown 2.1.5 and Vessel Cooldown 2.1.7, Revised 04/01/2011
Simulator JITT Reactor Shutdown 2.1.5 and Vessel Cooldown 2.1.7, Revised 02/19/2011
Training Schedules for Outage Training Cycle 03/14/2011 – 04/07/2011
Training Schedules for Training Cycle 02 02/13/2011 – 02/17/2011
Training Schedules for Training Cycle 01 11/22/2010 – 01/22/2011
Training Records and Remediation Training for Current Licensed Operators
Initial License Class 2009-2011 Class Schedule
O-RO-03-02, "Reactor Plant Startup Certification Unit Guide," Revision 10
O-RO-03-01-19, "Reactivity Management and Control Instructor/Student Guide," Revision 2
O-RO-03-01-20, "Simulator Scenario, Operations Standards," Revision 0
O-RO-03-02-01, "Instructional Module – Day One Cold Reactor Startup," Revision 7
O-RO-03-02-02, "Instructional Module – Day Two Hot Reactor Startup," Revision 7
O-RO-03-02-03, "Instructional Module – Day Three Cold Reactor Startup," Revision 3
O-RO-03-02-04, "Instructional Module – Day Four Hot Reactor Startup," Revision 3
O-RO-03-02-05, "Instructional Module – Day Five Cold Reactor Startup," Revision 3
O-RO-03-02-06, "Instructional Module – Day Six Cold Reactor Startup," Revision 3
O-RO-03-02-07, "Instructional Module – Day Seven 905 Certification Practice," Revision 3
O-RO-03-02-08, "Instructional Module – Day Eight 905 Certification Practice," Revision 2
O-RO-03-02-09, "Instructional Module – Day Nine Reactor Power Operations," Revision 1
O-RO-03-02-51, "Instructional Module – SOER 90-3 Nuclear Instrument Miscalibration,"
Revision 3

Miscellaneous:

Crew Briefing Sheet from May 10, 2011 SCRAM
Operations Section Standing Order 11-03
OSRC Meeting 2011-008 Meeting Minutes
Post-Trip Review Package from May 10, 2011 SCRAM with Attachments and Supporting Data
"EN-OP-116 Attachment 9.3 ITPE Supplemental Controls," developed for Post-Refueling
Outage Startup
Reactor Engineer's calculations pertaining to criticality prior to the reactor SCRAM
eSOMS Control Room Logs from 05/09/2011 through 05/11/2011
SRM and Moderator Temperature Traces with Calculated SRM Period 05/10/2011
Control Room Personnel Chart Dayshift 05/10/2011
Control Rod Notch History from Reactor Critical to Reactor SCRAM 05/10/2011
Control Rod Notch Worth Calculations for 05/10/2011 Reactor Startup
Power Maneuver Plan Cycle 19-01

SUPPLEMENTAL INFORMATION

LIST OF ACRONYMS

ACRS	Assistant Control Room Supervisor
ADAMS	Agency-wide Documents Access and Management System
AOM	Assistant Operations Manager
ATC	At the Controls
AV	Apparent Violation
BOP	Balance of Plant
CCDP	Conditional Core Damage Probability
CFR	Code of Federal Regulations
CR	Condition Report
CRD	Control Rod Drive
CRS	Control Room Supervisor
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
FFD	Fitness for Duty
HEP	Human Error Probability
HPCI	High Pressure Coolant Injection
HUR	Heat-up Rate
IMC	Inspection Manual Chapter
IPTE	Infrequently Performed Tests or Evolutions
IRM	Intermediate Range Monitor
JITT	Just in Time Training
NRC	Nuclear Regulatory Commission
OPS MGR	Operations Manager
PARS	Publicly Available Records
PD	Performance Deficiency
PNPS	Pilgrim Nuclear Power Station
POAH	Point of Adding Heat
PPC	Plant Process Computer
PRA	Probabilistic Risk Assessment
RCE	Root Cause Evaluation
RCIC	Reactor Core Isolation Cooling
RE	Reactor Engineer
RG	Regulatory Guide
RO	Reactor Operator
RO-ATC	Reactor Operator at the Controls
RPS	Reactor Protection System
SDP	Significance Determination Process
SM	Shift Manager
SRI	Senior Resident Inspector
SRM	Source Range Monitor
SRO	Senior Reactor Operator
SIT	Special Inspection Team
STA	Shift Technical Advisor
TS	Technical Specification

SPECIAL INSPECTION TEAM CHARTER



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

SPECIAL INSPECTION TEAM CHARTER

May 13, 2011

MEMORANDUM TO: Samuel L. Hansell Jr., Manager
Special Inspection Team

Raymond R. McKinley, Leader
Special Inspection Team

FROM: Christopher G. Miller, Director /RA/
Division of Reactor Safety

Darrell J. Roberts, Director /RA by Paul Krohn Acting For/
Division of Reactor Projects

SUBJECT: SPECIAL INSPECTION TEAM CHARTER -
PILGRIM NUCLEAR POWER STATION OPERATOR
PERFORMANCE DURING REACTOR STARTUP
ON MAY 10, 2011

In accordance with Inspection Manual Chapter (IMC) 0309, "Reactive Inspection Decision Basis for Reactors," a Special Inspection Team (SIT) is being chartered to evaluate operator performance and organizational decision-making associated with a reactor scram that occurred during a startup on May 10, 2011. The decision to conduct this special inspection was based on meeting the deterministic criteria (the event involved questions or concerns pertaining to licensee operational performance) and risk criteria specified in Enclosure 1 of IMC 0309. The calculable increase in conditional core damage probability (CCDP), which was in the low E-6 range, was based on application of an Initiating Event Analysis in Sapphire 8 due to the reactor scram, which was then modified for the conditions of the reactor when the transient occurred.

The SIT will expand on the event follow-up inspection activities started by the resident inspectors and augmented by a Division of Reactor Projects (DRP) inspector who was dispatched to the site soon after the event. The Team will review the causes of the event, and Entergy's organizational and operator response during and after the event. The Team will

SPECIAL INSPECTION TEAM CHARTER

perform interviews, as necessary, to understand the scope of operator actions performed during the event. The Team will also assess whether the SIT should be upgraded to an Augmented Inspection Team in accordance with IMC 0309.

The inspection will be conducted in accordance with the guidance contained in NRC Inspection Procedure 93812, "Special Inspection," and an inspection report will be issued within 45 days following the final exit meeting for the inspection.

The Special Inspection will commence on May 16, 2011. The following personnel have been assigned to this effort:

- Manager: Samuel L. Hansell, Jr., Branch Chief
Operations Branch, DRS, Region I
- Team Leader: Raymond R. McKinley, Senior Emergency Response Coordinator
Plant Support Branch, DRS, Region I
- Team Members: Brian C. Haagensen, Millstone Power Station Resident Inspector
Division of Reactor Projects, DRP, Region I
- David L. Molteni, Operations Engineer
Operations Branch, DRS, Region I

Enclosure: Special Inspection Team Charter

SPECIAL INSPECTION TEAM CHARTER

Special Inspection Team Charter Pilgrim Nuclear Power Station Operator Performance During Reactor Startup May 10, 2011

Background:

During startup from a refueling outage, Entergy operators withdrew rods to criticality the afternoon of May 10, 2011 and continued to withdraw control rods to the point of adding heat (approximately 1% power). While continuing to increase power, operators identified a higher than expected heat-up rate (HUR) with a five minute average HUR that, if allowed to continue, would have resulted in exceeding the technical specification limit. Operators made the Control Room Supervisor (CRS) and Shift Manager (SM) aware of the condition and proceeded to insert five control rods (two notches each) to lower the HUR to approximately 65°F/hr. At the time, it was not identified by the operators, reactor engineers or management oversight in the control room that the control rod insertions brought the reactor to a subcritical state (approximately 0.35% subcritical by later calculations). After reducing the HUR, the operators (without recognition of the subcritical reactor condition), proceeded to withdraw the five control rods back to their previous position. While withdrawing the fifth control rod back to its original position, the reactor experienced a full SCRAM on Intermediate Range Monitor (IRM) HI-HI flux signals. All rods inserted and equipment responded as expected.

Pilgrim initially investigated potential equipment related causes for the automatic scram as communicated to the NRC on the afternoon of May 10, 2011. Subsequent analysis revealed that human performance errors made by the operators were the cause of the scram. NRC was informed of this in the early morning hours of May 11, 2011. Entergy is continuing its investigation of the operator actions taken during this event. Entergy suspended the qualifications of the operators and the Shift Manager directly involved with the event while the investigation continues. Additional actions have been taken by Entergy that include more restrictive controls on reactivity additions following a negative reactivity insertion of any kind, briefing to other operating crews regarding the event, and initiation of a root cause evaluation. The Pilgrim resident inspectors and a resident inspector from a different site provided follow-up to this event under the Reactor Oversight Process (ROP) baseline inspection program.

Basis for the Formation of the SIT:

The IMC 0309 review concluded that one of the deterministic criteria was met due to questions or concerns pertaining to licensee operational performance. This criterion was met based on human performance errors that occurred and led to the unanticipated automatic reactor scram. The human performance errors included:

- Reactor operators were focused on monitoring heat-up rate (HUR) without appropriate focus on power level throughout the startup event;
- Reactor operators and control room supervision did not have proper sensitivity for the impacts from negative reactivity insertions with the reactor at low power conditions;

SPECIAL INSPECTION TEAM CHARTER

- The operators did not identify or utilize available plant indications that indicated the reactor was subcritical;
- Reactor operators did not follow shift manager instructions to maintain reactor power within the current IRM power band while addressing the elevated HUR;
- Operators and control room supervision did not engage reactor engineering staff with regard to planned rod movement after the reactor was made subcritical; and
- Prior to the identification of the unexpected HUR, reactor operators did not implement/enter the required abnormal operating procedure for a mispositioned control rod (Rod 30-11).

In accordance with IMC 0309, the event was evaluated for risk significance because one deterministic criterion was met. A Region I SRA evaluated the transient (reactor scram) from low reactor power using the Initiating Event Assessment feature of Sapphire 8. The IE-Trans basic event probability was set to 1.0 and all other initiating events were set to zero. The resulting dominant core damage sequences were subsequently evaluated by the SRA to account for the low reactor power conditions and alternating current (AC) power being supplied by off-site sources at the time of the event. The resulting conditional core damage probability (CCDP) was conservatively estimated in the low E-6 range, which is the overlap region between an SIT and No Additional inspection required. The dominant core damage sequences involve failure of direct current (DC) power sources and failure of residual heat removal. However, with the low decay heat load following the refuel outage, these core damage sequences represent a conservative estimate of risk.

Additionally, this event involved multiple licensed operators not recognizing the reactivity status of an operating reactor during startup and demonstrating a poor understanding of reactor physics in a low power condition. In light of the aforementioned human performance errors, and consistent with the risk evaluation and Section 4.04, Region I has decided to initiate an SIT.

Objectives of the Special Inspection:

The Team will review the causes of the event, and Entergy's organizational and operator response during and following the event. The Team will perform interviews, as necessary, to understand the scope of operator actions performed during the event.

To accomplish these objectives, the Team will:

1. Develop a complete sequence of events including follow-up actions taken by Entergy, and the sequence of communications within Entergy and to the NRC subsequent to the event;
2. Review and assess crew operator performance and crew decision making, including adherence to expected roles and responsibilities, the use of the command and control elements associated with reactivity manipulations, the use of procedures, the use of diverse instrumentation to assess plant conditions, response to alarms and overall implementation of operations department and station standards;

SPECIAL INSPECTION TEAM CHARTER

3. Evaluate the extent of condition with respect to the other crews;
4. Review the adequacy of operator requalification training as it relates to this event, including the integration of newly licensed operators into the operator requalification training program;
5. Review the adequacy of the preparation by the operations staff for the reactor startup including training prior to the evolution and briefings by the operations staff.
6. Review the adequacy of the simulator to model the behavior of the current reactor core during startup activities and the current adequacy of the simulator for use in reactor startup training;
7. Assess the decision making and actions taken by the operators and station management during the initial and subsequent reactor startup to determine if there are any implications related to safety culture;
8. Review and assess the effectiveness of Entergy's response to this event and corrective actions taken to date. This includes overall organizational response, and adequacy of immediate, interim and proposed long-term corrective actions. This will also include evaluation of the root cause analysis when developed by the licensee;
9. Review the adequacy of the Entergy and Site fitness for duty processes and procedures when a human performance error has occurred;
10. Evaluate Entergy's application of pertinent industry operating experience, including INPO SOER 10-2, "Engaged, Thinking Organizations," INPO SOER 07-1, "Reactivity Management," and other recent events involving reactivity management errors to assess the effectiveness of any actions taken in response to the operating experience; and
11. Document the inspection findings and conclusions in a Special Inspection Team final report within 45 days of inspection completion.

Guidance:

Inspection Procedure 93812, "Special Inspection", provides additional guidance to be used by the SIT. Team duties will be as described in Inspection Procedure 93812. The inspection should emphasize fact-finding in its review of the circumstances surrounding the event. Safety concerns identified that are not directly related to the event should be reported to the Region I office for appropriate action.

The Team will conduct an entrance meeting and begin the inspection on May 16, 2011. While on-site, the Team Leader will provide daily briefings to Region I management, who will coordinate with the Office of Nuclear Reactor Regulation to ensure that all other pertinent parties are kept informed. The Team will also coordinate with the Region I State Liaison Officer

SPECIAL INSPECTION TEAM CHARTER

to implement the Memorandum of Understanding between the NRC and the State of Massachusetts to offer observation of the inspection by representatives of the state. A report documenting the results of the inspection will be issued within 45 days following the final exit meeting for the inspection.

Before the end of the first day onsite, the Team Manager shall provide a recommendation to the Regional Administrator as to whether the SIT should continue or be upgraded to an Augmented Inspection Team response.

This Charter may be modified should the Team develop significant new information that warrants review.

DETAILED SEQUENCE OF EVENTS**May 10, 2011, Reactor Scram Event**

The team constructed the sequence of events from a review of control room narrative logs, plant process computer (PPC) data (alarm printout, sequence of event printout, plant parameter graphs) and plant personnel interviews.

Time	Event
05/09/11	
Two Sessions	Just In Time Training (JITT) was conducted for the reactor startup. Certain key members of the operating crew that were directly involved with this event were not present for the training including the Shift Manager (SM), the Assistant Control Room Supervisor (ACRS) who temporarily relieved the Control Room Supervisor (CRS) prior to the scram, and the Reactor Operator who was at the controls (RO-ATC) when the scram occurred.
05/10/11	
0626	The reactor mode switch was moved to the startup position.
~0630	The oncoming day shift operators received a reactor maneuvering plan briefing. The Reactor Engineers (REs) led the brief.
0641	Operators commenced control rod withdrawal.
0700	The day shift operating crew assumed the shift, and control rod withdraw continues.
1212	The reactor became critical.
1227	The point of adding heat was reached.
~1231	The CRS was relieved for lunch by the ACRS. The oncoming CRS providing the relief did not receive Just In Time Training (JITT), nor did he participate in the reactor maneuvering plan briefing.
~1231	The RO-ATC was relieved for lunch by the Licensed Operator previously assigned as the ATC verifier. The oncoming RO-ATC providing the relief did not receive Just In Time Training (JITT), but he did participate in the reactor maneuvering plan briefing.
~1231	A Licensed Operator previously assigned to other startup activities was reassigned to fill the role of ATC verifier. This individual received JITT training, and he also received a separate reactor maneuvering plan briefing from a RE upon arriving to work at approximately 1100.
1246	The RO-ATC withdrew 5 rods 2 notches to establish a heat-up rate.

DETAILED SEQUENCE OF EVENTS

Time	Event
1255	The RO-ATC attempts to withdraw control rod 30-11 from position 08 to position 10 using single notch control, but the control rod does not move. The RO-ATC raised drive water pressure and attempted several notch withdraw commands, but the control rod failed to move.
1257	The RO-ATC attempts to withdraw control rod 30-11 from position 08 to position 10 using a "double clutch" maneuver, but the control rod incorrectly inserted one notch to position 06. The RO-ATC does not discuss the control rod mispositioning error with the crew.
1257	The ATC verifier and CRS also saw control rod 30-11 move incorrectly to position 06, but the control rod mispositioning error is not discussed.
1302	The RO-ATC then withdraws control rod 30-11 from position 06 to position 12.
~1305	The crew observes that the 5 minute average reactor coolant heat-up rate is 18°F over the 5 minute period, and the crew determines that this corresponded to a 216°F/hour heat-up rate. In actuality, the 5 minute average heat-up rate reflected the instantaneous heat-up rate. The actual hourly heat-up rate was 50°F/hour. The crew informs the SM of the perceived heat-up rate.
~1306	The SM directed the RO-ATC to insert control rods to reduce the heat-up rate, but the SM did not specify the number of control rods or notches to insert.
1307	The RO-ATC begins to drive 5 rods 2 notches into the core to the reduce heat-up rate.
~1308	The REs question the SM regarding the decision to insert control rods, and the SM told the REs that the insertion was needed to control the heat-up rate. There was no further discussion.
~1309	The Assistant Operations Manager (AOM-Shift) cautioned the SM that there was the potential to drive the reactor sub-critical by inserting control rods and that they needed to be careful. The SM also recalled being concerned about the potential to drive the reactor sub-critical. The operating crew at the controls was not made aware of these concerns.
1310	Control rod insertion is stopped. The control rods are now at the same position as when the reactor initially became critical; however, moderator temperature is now 40°F higher than it was at initial criticality. The higher moderator temperature in conjunction with the control rod insertion rendered the reactor sub-critical, but the operators were not aware of this.
~1310	The SM left the control room to take a break, and the AOM-Shift left the controls area to get his lunch in the control room kitchen.

DETAILED SEQUENCE OF EVENTS

Time	Event
~1311	The operators range down the Intermediate Range Monitors (IRMs) two decades from Range 8 to Range 6 in response to the lowering neutron flux.
~1312	The original CRS returns from break and resumes duties as CRS as well as responsibility for the reactivity maneuver as the Reactivity SRO.
1313	After observing a 0°F/hour heat-up rate, the CRS directs the RO-ATC to resume control rod withdrawal to establish a positive heat-up rate. The RO-ATC begins to withdraw 5 rods 2 notches each to restore the heat-up rate.
1315	While notch withdrawing control rod 14-19 from position 08 to position 12, IRM readings begin to rise again requiring the operators to range up on the IRMs in response to the rising neutron flux. The reactor has returned to a critical condition, but the operators are not aware of the change in reactor status with regards to criticality.
1316	The RO-ATC notch withdraws control rod 22-43 from position 08 to position 12 resulting in a more rapid rise in IRM readings. The reactor period was calculated to be 40 seconds during the post trip review.
~1318	The RO-ATC attempts to notch withdraw control rod 30-11 from position 08 to position 10 resulting in a sharp rise in IRM readings.
1318	The reactor automatically scrammed on IRM high-high flux level prior to completing the withdrawal of rod 30-11 to position 10. Post event analysis determined that the reactor period was approximately 20 seconds, and that the scram occurred at approximately 1.7% equivalent Average Power Range Monitor (APRM) power.
~1320	The RE stated that he recognized that the operators had caused the reactor scram by withdrawing rods to criticality.
1345	The crew debriefed the events leading up to the reactor scram.
~1400	The RE participated in a conference call with the fuels group in Jackson (corporate reactor engineering staff) to discuss the event. The RE informed the conference call participants that the reactor scram had been caused by human error.
~1600	The RE participated in a conference call with General Electric (GE) to discuss the event.
~1630	The RE informed the Director of Engineering that the reactor scram was caused by human error.
~1700	The RE informed the General Manager Plant Operations (GMPO) that the reactor scram was caused by human error. The GMPO asked the RE to draft a memo describing what happened and send it to him.

DETAILED SEQUENCE OF EVENTS

Time	Event
1730	The GMPO met with the Operations Manager (OPS MGR) and the operators involved in the re-criticality to discuss the events.
~1900	After shift turnover, the Assistant Operations Manager (AOM) recognized that human error was the cause of the scram. Equipment issues had been ruled out.
~1930 To ~2200	The GMPO recalls meeting with the OPS MGR, RE and corporate core design group to discuss issues associated with the scram. The GMPO indicated that his team was certain that the scram was caused by a human performance / knowledge deficiency problem.
~2330	The Operations Manager (OPS MGR) prepared a written briefing for the crew on the event.
5/11/11	
0030	An On-site Safety Review Committee (OSRC) conference call was convened to review the event and evaluate a recommendation to restart the reactor.
0130	The OSRC recommended restarting the reactor. The GMPO was briefed regarding the OSRC recommendations.
0200	The GMPO approved restarting the reactor. He directed the OPS MGR to call the NRC Senior Resident Inspector (SRI).
0200	The OPS MGR called the SRI to inform him of the decision to restart the plant. The OPS MGR informed the SRI that the cause of the scram was due to human error.
0215	The SRI called the NRC Region I Division of Reactor Projects (DRP) Branch Chief to inform him of the decision to restart the plant. The SRI then responded to the site to observe the startup.
~0300	The reactor mode switch was placed in the startup position.
~0300	The SRI arrives onsite.
~0300	The DRP Branch Chief called the GMPO to discuss the decision to restart the reactor.

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Qualitative Decision-Making Attributes for NRC Management Review

Decision Attribute	Applicable to Decision?	Basis for Input to Decision – Provide qualitative and/or quantitative information for management review and decision making.
Finding can be bounded using qualitative and/or quantitative information?	No	<p>IMC 0609 Appendix G is not appropriate since the conditions for reactor shutdown operations were not met. The at-power safety Significance Determination Process, IMC 0609 Appendix A, quantitative analysis methodology is not adequate to provide reasonable estimates of the finding's significance. Furthermore, the SDP does not model errors of commission and does not provide a method of accurately estimating changes to the human error probabilities caused for errors of omission. As a result, no quantitative risk evaluation can be performed for this finding.</p> <p>Improper use and execution of procedures coupled with weak work control practices has the potential to increase the human error probability (HEP) for credited operator actions. The probabilistic risk assessment models are highly sensitive to small variations in HEP changes. The existing PRA research does not currently support a method for varying the performance shaping factors in response to defined error forcing contexts. It is not possible to calculate a valid single point risk estimate. Human performance is a very large contributor to PRA uncertainty.</p>
Defense-in-Depth affected?	Yes	<p>The term "defense in depth" is commonly associated with the maintenance of the integrity and independence of the three fission product barriers as well as emergency response actions. In addition, redundant and diverse safety systems, including trained licensed operators conducting operations in accordance with approved station procedures that were developed under an approved quality control program are integral to maintaining a "defense in depth." While an automatic reactor scram was initiated as designed to protect the core during this event, the fuel barrier was not actually compromised by the crew's actions since the automatic protective action was successful.</p> <p>However, this performance deficiency revealed organizational and human performance weaknesses which eroded defense in depth. The operating crew</p>

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		<p>plays a vital role in the maintenance of "defense in depth" from the perspective that they directly operate station controls. Human errors can lead to consequences that have the potential to compromise the three fission product barriers. The commission of multiple unforeseen human errors in a short period of time during the reactor startup degraded the operator's performance as an important "defense in depth" barrier. These operator human performance errors resulted in a challenge to the automatic Reactor Protection System which successfully terminated the event in this particular case.</p>
<p>Performance Deficiency effect on the Safety Margin maintained?</p>	<p>Yes</p>	<p>This performance deficiency had the potential to adversely affect the margin of safety. In this particular event, the failure to implement conduct of operations and reactivity control standards and procedures led to a reactor protection set-point being exceeded, causing a reactor scram. In fact, non-conservative operator actions led to an unrecognized subcriticality followed by an unrecognized return to criticality. These operator actions caused a rapid rise in neutron flux and reactor power such that the IRM HI-HI neutron flux reactor trip set point was exceeded resulting in an automatic reactor scram.</p> <p>In this case, the IRM HI-HI neutron flux RPS protective function successfully terminated the event and prevented exceeding fuel barrier design safety margin and the potential for subsequent fuel barrier damage. It should also be noted that the Average Power Range Monitor (APRM) Low Power RPS set point was available as a backup to the IRM trip function. The APRM Low Power set point will initiate a reactor scram at less than or equal to 15% power whenever the mode switch is NOT in "RUN".</p> <p>While there was no reduction in the quantitative design margin, there was a qualitative reduction in the safety margin as there is an expectation that the operators will maintain an understanding of the status of the reactor and approach criticality in a deliberate and carefully controlled manner. In this case, the operators lost situational awareness regarding the status of the reactor and subsequently initiated incorrect actions that led to an unrecognized subcriticality followed by an</p>

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		unrecognized return to criticality resulting in an automatic reactor scram.
The extent the performance deficiency affects other equipment.	Yes	The inspectors reviewed the Entergy root cause evaluation team report and determined that the underlying causes of this performance deficiency exist across the Operations organization. This includes weaknesses in oversight, human performance behaviors, as well as operator knowledge, skills, and abilities deficiencies associated with low power reactor physics and operations in the IRM range. It should be noted that the performance deficiency did not degrade physical plant equipment; however, the requirement that licensed operators conduct licensed activities in accordance with station approved procedures is integral to maintaining plant safety. Faulty operator performance has the potential to adversely affect plant equipment.
Degree of degradation of failed or unavailable component(s).	N/A	N/A
Period of time (exposure time) effect on the performance deficiency.	Yes	<p>With respect to the issues underlying this performance deficiency, the exposure time is indeterminate, but clearly developed over an extended period of time.</p> <p>The Entergy root cause evaluation team determined that the causal factors for the event had existed for a considerable period of time, but they did not quantify the exposure time. A number of condition reports were written over the last year, including a Fleet Assessment performed in February 2011, which identified shortfalls in oversight and adherence to conduct of operations human performance standards.</p> <p>This assessment is complicated by the fact that there were not any apparent significant licensed operator performance issues at Pilgrim before this event. In the Human Performance cross-cutting area, none of the aspects currently has a theme, nor has there been a theme in the recent past. The behaviors outlined by the performance deficiency have not been observed by the resident inspector staff prior to this event.</p>

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<p>The likelihood that the licensee's recovery actions would successfully mitigate the performance deficiency.</p>	<p>Yes</p>	<p>Although "recovery actions" do not equate to "corrective actions," this section lends itself to a discussion of licensee corrective action in that completion of these actions would mitigate the performance deficiency.</p> <p>The licensee's root cause analysis was thorough and appeared to identify all underlying causal factors. The associated proposed corrective actions appear to adequately address the underlying causal factors. Short term corrective actions have been completed to correct the specific issues associated with this event. Longer term corrective actions are in progress to address programmatic weakness in training and human performance behaviors.</p>
<p>Additional qualitative circumstances associated with the finding that regional management should consider in the evaluation process.</p>	<p>Yes</p>	<p>In this event, there were a significant number of lapses in operator human performance fundamentals as described in the conduct of operations and reactivity control standards and procedures. These lapses in human performance fundamentals degraded individual operator performance, crew performance, as well as management oversight performance. The lack of enforcement of, and adherence to, the conduct of operations and reactivity control standards and procedures were identified as the root cause of the reactor scram event.</p> <p>The inspectors, as well as the Entergy root cause evaluation team, determined that the extent of condition existed across multiple crews of the Operations department and has the potential to exist across all Pilgrim Nuclear Power Station departments.</p> <p>It should be noted that overall licensee operational performance has been acceptable. The plant runs well, and there are few challenges to the licensed operators since the plant tends to run reliably through the operating cycle.</p> <p>The inspectors noted that licensee corrective actions to correct this performance deficiency prior to this event were ineffective, and that this pattern continued to manifest itself immediately before the reactor scram and in the days immediately following the reactor scram. For example, the Entergy root cause team identified a number of condition reports that were</p>

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	<p>written over the past year that identified shortfalls in oversight and adherence to conduct of operations human performance standards. Corrective actions were narrowly focused and failed to arrest the degrading trend. Inspectors also noted that, during the startup leading to the reactor scram, there were numerous lapses in human performance fundamentals and missed opportunities to correct those behavioral deficiencies. Immediately following the reactor scram, the licensee's post trip reviews and OSRC reviews failed to fully evaluate the extent and scope of the human performance and knowledge deficiencies prior to authorizing the restart of the reactor. For instance, NRC inspectors identified that a control rod had been mispositioned during the startup and that an <i>Infrequently Performed Test or Evolution (IPTE)</i> briefing had not been conducted during the initial and subsequent startups. The control rod mispositioning and failure to perform the IPTE briefing were not identified by the licensee. In addition, in the days immediately following the event, inspectors continued to observe a lack of formality in operator communications, a lack of apparent peer checking, and a number of control room distractions.</p> <p>While it will clearly take time to fully change the behaviors associated with this performance deficiency, the inspectors did observe progress being made during the inspection. The licensee's Significant Event Review Team (SERT) and root cause analysis team performed thorough reviews of the event, and the licensee has identified a number of appropriate corrective actions that should correct the performance deficiency. In addition, licensee line personnel up through senior plant management were interviewed extensively by the inspectors in the days and weeks following the event, and it appears as though the licensee has fully internalized the significance of this event.</p> <p>However, while progress is being made to correct the performance deficiency, additional follow-up inspection(s) may be warranted to confirm the future effectiveness of the licensee's corrective actions.</p>
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