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.

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)

cc w/encl: Distribution via ListServ

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,

/RA/

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)

cc w/encl: Distribution via ListServ

Distribution: See next page

SUNSI Review Complete: rrm* (Reviewer's Initials)
DOCUMENT NAME: G:\DRS\Operations Branch\MCKINLEY\Pilgrim SIT June 2011\Inspection Report Drafts\Pilgrim SIT Concurrence\Pilgrim 2011 SIT Report Final.docx
After declaring this document "An Official Agency Record" it **will** be released to the Public. **ML112440100**
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	RI/DRS	RI/DRS	RI/ORA	RI/DRP	RI/DRP
NAME	RMcKinley/rrm* Prior concurrence	DJackson/dej* Prior concurrence	DHolody/aed for* Prior concurrence	RBellamy/tcs for* Prior concurrence	DRoberts/djr* Prior concurrence
DATE	08/19/11	08/19/11	08/19/11	08/ /11	08/30/11

OFFICE	RI/DRS				
NAME	CMiller/cgm				
DATE	08/31/11				

OFFICIAL RECORD COPY

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

Enclosure

TABLE OF CONTENTS

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 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. <u>Operator Human Performance</u>

a. <u>Inspection Scope</u>

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.

- 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.