Entergy

**Entergy Nuclear Generation Co.**
Pilgrim Nuclear Power Station
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**Mike Bellamy**
Site Vice President

10 CFR 50.73

July 16, 2001
ENGC Ltr. 2.01.074

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

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

Dear Sir:

The enclosed Licensee Event Report (LER) 2001-005-00, "Manual Scram While Subcritical Due to Personnel Error," is submitted in accordance with 10 CFR 50.73.

This letter contains no commitments.

Please do not hesitate to contact me if there are any questions regarding this report.

Sincerely,

Mike Bellamy

DWE/
Enclosure: LER 2001-005-00

cc: Mr. Hubert J. Miller
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Senior NRC Resident Inspector

Mr. Steven Bloom
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INPO Records

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NRC Form 366 U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

(See reverse for number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
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TITLE (4)

Manual Scram While Subcritical Due To Personnel Error

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAME N/A | DOCKET NUMBER 05000 |
| 05 | 16 | 2001 | 2001 | 005 | 00 | 07 | 16 | 2001 | FACILITY NAME N/A | DOCKET NUMBER 05000 |

| OPERATING MODE (9) | N | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | 20.2201 (b) | | 20.2203(a)(2)(v) | | 50.73(a)(2)(i)(B) | | 50.73(a)(2)(viii) |
| POWER LEVEL (10) | 000 | | 22.2203(a)(1) | | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii)(B) | | 50.73(a)(2)(x) |
| | | | 20.2203(a)(2)(i) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(iii) | | 73.71 |
| | | | 20.2203(a)(2)(ii) | | 20.2203(a)(4) | X | 50.73(a)(2)(iv) | | OTHER |
| | | | 20.2203(a)(2)(iii) | | 50.36(c)(1) | | 50.73(a)(2)(v)(D) | | Specify in Abstract below |
| | | | 20.2203(a)(2)(iv) | | 50.36(c)(2) | | 50.73(a)(2)(vii)(D) | | or in NRC Form 366A |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Bryan S. Ford – Licensing Manager | (508) 830-8403 |

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

| CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANUFACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | |
| | | | | | | | | | |

| SUPPLEMENTAL REPORT EXPECTED (14) | | | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|---|
| | YES (If yes, complete EXPECTED SUBMISSION DATE) | X | NO | | | | |

**ABSTRACT** (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On May 16, 2001 at 1557 hours, a reactor protection system scram signal was manually initiated due to increasing reactor water level. The event occurred during a plant startup with the reactor in a sub-critical condition. The signal resulted in the insertion of all control rods that were withdrawn at the time of the event.

The root cause was the failure to follow procedure. A test engineer changed the preferred lineup for a leak test of a valve. The lineup resulted in the isolation of the reactor water cleanup system due to air introduced into the system during the test. The isolation resulted in an increase in reactor water level. Corrective action taken included satisfactory completion of the leak test. Training and test procedure revision are planned.

The event posed no threat to public health and safety.

BACKGROUND

On May 16, 2001, Pilgrim Station startup activities were underway from the 2001 refueling outage (RFO-13).

The reactor was sub-critical with the reactor mode selector switch in the STARTUP position. Control rods were being withdrawn in accordance with the startup procedure. The reactor water level was normal at about +30 inches (narrow range level). The reactor vessel pressure was about 8 psig and the reactor water temperature was about 215 degrees Fahrenheit.

The Reactor Water Cleanup (RWCU) System was in service. Excess reactor water inventory was being rejected to the Main Condenser via the RWCU System reject valve. The outboard Sample System isolation valve, AO-220-45, was closed for a post work leak rate test of the valve. The in-series isolation valve was also closed. At about 1430 hours leak rate testing activities began for valve AO-220-45.

At 1444 hours, a RWCU System isolation signal occurred. The signal was the result of a high temperature condition at the outlet of the RWCU non-regenerative heat exchanger. The signal is non-safety related and functions to protect the RWCU demineralizer media. The signal resulted in the automatic closing of a RWCU suction isolation valve located in the piping from the reactor vessel and trip of the RWCU pump that was in service. The closing of the isolation valve eliminates the RWCU System as a means to reject excess reactor water inventory and consequently, the water level began to increase because of the addition of water into the reactor vessel from the Control Rod Drive (CRD) System.

The control room licensed operators started following the actions of Procedure 2.4.27, "Reactor Water Cleanup System Malfunctions," in response to the isolation. The in-series RWCU suction isolation valve was manually closed. CRD System cooling water flow was minimized to about 30 gpm.

The licensed operators held a control room briefing for initiating a manual scram if reactor water level increased to +65 inches (narrow range level). In accordance with the briefing, the main turbine was tripped, at 1500 hours, and the main steam isolation valves were closed via control switches in the main control room, at 1515 hours, due to increasing reactor water level.

After the initial investigation into the course of the isolation and system venting, the RWCU isolation valves were reopened at 1530 hours. The RWCU isolation valves again closed due to a high temperature at the outlet of the non-regenerative heat exchanger at 1543 hours.

EVENT DESCRIPTION

On May 16, 2001 at 1557 hours, a manually initiated reactor protection system scram signal and scram occurred. The scram signal was initiated by the movement of the reactor mode switch to the SHUTDOWN position.

The event occurred during a startup with the reactor in a sub-critical condition, and with the reactor water level at about +54 inches (narrow range level). The scram signal resulted in the insertion of the control rods that were withdrawn at the time of the event. All control rods inserted and were verified fully inserted.

After investigation and system venting, the RWCU System was returned to service at 1715 hours.

A corrective action program document (PR 01.9486) was written to document the event. The NRC Operations Center was notified of the event in accordance with 10 CFR 50.72(b)(3)(iv)(A) at 1916 hours on May 16, 2001.

CAUSE

The cause of the scram signal was the intentional licensed operator movement of the reactor mode selector switch from the STARTUP position to the SHUTDOWN position. This intentional action was taken in response to increasing reactor water level. The increase in water level was due to the isolation of the RWCU System and water addition into the reactor vessel from the Control Rod Drive System. The isolation was the result of air introduced into the RWCU System during the leak rate test of valve AO-220-45. The air reduced the heat transfer in the RWCU heat exchangers and resulted in the high temperature isolations including the isolation that resulted in the event.

The root cause of the event was the failure to follow the test procedure. A utility non-licensed test engineer did not properly establish the test vent flowpath. Contributing causes included less than adequate safeguards in the procedure for making a change to a pre-established test lineup.

The root cause analysis report was being finalized when this report was submitted. This report will be supplemented if the finalized root cause analysis identifies significant new information.

CORRECTIVE ACTION

The RWCU System was vented and returned to service on May 16, 2001. The leak rate test of the outboard sample isolation valve, AO-220-45, was satisfactorily completed on May 17, 2001.

Corrective actions planned include:

- Including this event in the training program for test engineers.
- Revising the leak rate test procedure to ensure appropriate reviews exist when a test engineer makes changes to a pre-approved test lineup.

The completion of the root cause analysis and actions to be taken will be tracked in the Pilgrim Station Corrective Action Program (PR 01.9486).

SAFETY CONSEQUENCES

The event posed no threat to public health and safety.

The RWCU System is non-safety related. As part of the primary containment system, the RWCU isolation valves are safety related and are designed to automatically close from any one of several trip signals including a high temperature condition sensed by instrumentation at the outlet of the RWCU non-regenerative heat exchanger. The high outlet temperature isolation is non-safety related and functions to protect the RWCU filter demineralizer media. For this event, the isolation interrupted the rejection of excess reactor water inventory and resulted in an increase in reactor water level, the manual closing of the MSIVs, and a manual scram (sub-critical).

All control rods that were withdrawn at the time of the event inserted into the core as designed. The availability of the low-pressure core cooling systems was not affected by the RWCU System isolations or closing of the main steam isolation valves.

REPORTABILITY

This report was submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the scram, although sub-critical, resulted from a valid signal (intentional operator actuation of the reactor protection system).

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station LERs for similarity to previous events. The review identified no similar event.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

| COMPONENTS | CODES |
|---|---|
| Exchanger, Heat | HX |
| Switch, Hand (RPS mode selector switch) | HS |
| Valve, isolation | ISV |

SYSTEMS

| | |
|---|---|
| Main Steam System | SB |
| Plant Protection System (RPS) | JC |
| Reactor Water Cleanup (RWCU) System | CE |
| Sampling and Water Quality System | KN |

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**TEXT** (If more space is required, use additional copies of NRC Form 366A) **(17)**