Entergy

**Entergy Nuclear Generation Co.**
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

**Mike Bellamy**
Site Vice President

February 25, 2002

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

SUBJECT: Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

Licensee Event Report 2001-007-00

LETTER NUMBER: 2.02.007

Dear Sir:

The enclosed Licensee Event Report (LER) 2001-007-00, "Automatic Scram During Transient Caused by Failure of Calibrating Unit," 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,

FOR RMB

Mike Bellamy

DWE/jbb

Enclosure: LER 2001-007-00

cc: Mr. Hubert J. Miller
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U.S. Nuclear Regulatory Commission
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NRC Form 366 (6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER)

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



| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE(3) |
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| PILGRIM NUCLEAR POWER STATION | 05000293 | 1 of 5 |

**TITLE (4)** Automatic Scram During Transient Caused by Failure of Calibrating Unit

| 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 |
| 12 | 27 | 2001 | 2001 | 007 | 00 | 02 | 25 | 2002 | 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) | 053 | | 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)(A) | | 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(3)(1)(ii)(A) | | 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 |
|---|---|---|---|---|---|---|---|---|---|
| X | JG | TD | R369 | Y | | | | | |
| | | | | | | | | | |

| 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 December 27, 2001 at 1321 hours, an automatic Reactor Protection System scram signal and scram occurred during a surveillance test performed while at 100 percent reactor power. The scram resulted in the automatic insertion of all control rods.

The direct cause of the scram was an automatic trip of both Recirculation System motor-generator sets that caused decreased core flow resulting in a trip signal from the neutron monitoring system flow control trip reference circuits. The root cause of the scram was an internal failure, during the required surveillance test, of an analog trip system calibrating unit that was not reproducible. Circuit analysis identified the most likely failed component within the unit was an intermittent micro-switch. A failed micro-switch would apply a voltage from the calibrating unit to a selected master trip unit while rotating the same rotary selector switch between channels. Corrective action taken included replacement of the suspect calibrating unit.

The event posed no threat to public heath and safety.

## BACKGROUND

The analog trip system consists of transmitters, master and slave trip units, power supplies, and calibrating units. The transmitters are located on instrument racks connected to the reactor vessel or drywell, and function to monitor parameters that include reactor vessel water level and pressure and Drywell pressure. The trip units are installed in instrument panels and receive signals from the transmitters. The trip units function to initiate signals to the related systems. The calibrating units are used for calibration of the trip units.

The anticipated transient without scram (ATWS) panels are part of the instrument channels for the alternate rod insertion function (ARI), feedwater pump trip function, and recirculation pump trip (RPT) function. The RPT functions to trip the circuit breakers of the motors that drive the reactor recirculation motor-generator (MG) sets. Each MG set provides power to the respective reactor recirculation pump. The recirculation pumps provide the motive force for reactor core flow. There are two ATWS divisions, Division 1 and Division 2. Each division is designed to fulfill the ATWS trip functions and is housed in a separate panel.

Technical Specifications 3.2.G and 4.2.G pertain to the operability and surveillance requirements for the ATWS ARI and RPT trip functions. The required surveillance test includes testing of the panel power supplies, calibration of the trip units, and functional tests of trip relays and indicating lights. The surveillance is performed in accordance with approved procedure.

On December 27, 2001, plant conditions included the following: The reactor was operating at 100 percent power. The reactor mode selector switch was in the RUN position. The reactor vessel pressure was approximately 1034 psig with the reactor water at the saturation temperature for that pressure. The reactor vessel water level was +30" (narrow range) and was being controlled in the three element control mode. Reactor core flow was in the loop control mode. Reactor core flow was approximately 55.5 E+06 pounds per hour. The ATWS Division 2 components were being surveillance tested in accordance with PNPS Procedure 8.M.1-29 (Rev. 34) Attachment 2.

After the satisfactory test of the Division 2 power supplies, calibrating unit preparation, calibrations of three master trip units and one slave trip unit, and functional tests of the related relays and indicating lights, master trip unit PIS-263-123D and the related relays and indicating lights were to be calibrated and functionally tested. After verifying the prerequisites and reviewing the precautions, the Division 2 calibrating unit "Press to Cal" knob (smaller knob) was pulled out and the larger knob was set to slot number 5 (PIS-263-123D) in accordance with the procedure. This action resulted in the unexpected trip of both reactor recirculation MG sets and consequent trip of the recirculation pumps.

## EVENT DESCRIPTION

On December 27, 2001 at 1321 hours, an automatic Reactor Protection System (RPS) scram signal and scram occurred during a transient while at approximately 53 percent reactor power. The scram was uncomplicated and resulted in the automatic insertion of the control rods, automatic transfer of source of the plant's electrical power system, automatic trip of the main turbine generator, and automatic opening of the applicable switchyard circuit breakers accordance with design.

As expected, the scram resulted in a decrease in the reactor vessel water level due to the decrease in the void fraction in the water. The decrease, to approximately +4.5" (narrow range level), resulted in the automatic actuation of the Primary Containment Isolation Control System (PCIS) and Reactor Building Isolation Control System (RBIS) in accordance with design. The PCIS actuation included the automatic closing of the Primary Containment System Group 2 isolation valves (sample valves) that were open and the closing of the Group 6 isolation valves (Reactor Water Cleanup System). The RBIS isolation signal resulted in the automatic closing of the Secondary Containment System/Reactor Building ventilation supply and exhaust dampers and automatic start of the Standby Gas Treatment System.

Utility licensed operators proceeded to take the actions to place the unit in a cold shutdown condition. The insertion of all control rods was verified.

The NRC Operations Center was notified in accordance with 10 CFR 50.72 on December 27, 2001 at 1603 hours.

During the cooldown to cold shutdown, at approximately 0330 hours on December 28, 2001, minor notching of reactor vessel water level instrumentation was observed on the "B" reactor water level instrumentation. No notching was observed on the other reactor vessel water level instrumentation. The notching was investigated and evaluated for impact on operability of the related instrumentation. The evaluation concluded the instrumentation was operable.

The Residual Heat Removal (RHR) System was put into service in the shutdown cooling mode at 0437 hours, with one RHR train "B" pump in operation. Cold shutdown conditions were achieved at 0550 hours.

A corrective action program document (PR 01.8151) was written to document the scram. PR 01.8152 was written to document the reactor water level instrumentation notching.

## CAUSE

The direct cause of the scram was an automatic trip of the circuit breakers that power the drive motors of both Recirculation System motor-generator sets. The trip was initiated by the ATWS/RPT circuitry. Testing of the ATWS/ARI circuitry after the event verified the ARI circuitry operated properly and an ARI trip did not occur. The ATWS/RPT trip resulted in decreased core flow that resulted in trip signals from the neutron monitoring system flow control trip reference circuits in accordance with design.

The root cause of the scram was an internal failure within the calibrating unit that was not reproducible. Circuit analysis and other investigative efforts identified the most likely failed component within the calibrating unit was an intermittent micro-switch. A failed micro-switch would apply 24V DC from the calibrating unit to a master trip card while rotating the same rotary selector switch between channels. This failure would bypass a system interlock designed to prevent selecting a channel while rotating the calibrating unit "press to cal" knob.

The suspect calibrating unit was functionally tested with satisfactory results before and after removal from the ATWS Division 2 panel. The calibrating unit was manufactured by Rosemount, Incorporated, model number 510 DU.

CORRECTIVE ACTION

Corrective actions taken included:

The suspect calibrating unit was replaced with a Rosemount model 710 DU calibrating unit that is equivalent to a model 510 DU calibrating unit.

The ATWS Division 2 panel was inspected with satisfactory results.

The ATWS Division 2 trip units were functionally tested with satisfactory results.

Corrective actions planned include:

Revision of applicable surveillance procedures. The focus of this action is to verify that no alarms and/or trip unit status light actuate and the calibrating unit "cal" light does not initiate during the manipulation of the rotary selector switch.

Evaluation of other analog trip system calibrating units.

OTHER ACTION TAKEN

The "B" reactor vessel water level instruments were vented and the related instrument rack was backflushed. Longer term actions are being tracked in the Corrective Action Program.

SAFETY CONSEQUENCES

Although the failure of the calibrating unit micro-switch resulted in an unnecessary plant transient, the event posed no threat to public health and safety.

The scram was the designed response to the high neutron flux trip signal that resulted from the trip of the Recirculation System motor-generator sets/pumps that occurred while at 100 percent power. The trip of the MG sets/pumps was the designed response to (false) trip signals from the ATWS Division 2 circuitry for the Recirculation Pump Trip function.

REPORTABILITY

This report was submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A) because the RPS scram signal and consequent PCIS and RBIS isolation signals were not planned.

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

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports submitted since 1987. The review focused on reports that involved ATWS or ATS and/or applicable procedure. The review identified no similarity.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

| COMPONENTS | CODES |
|---|---|
| Condensing unit (Reference leg "B") | CDU |
| Panel (ATWS Division 2) | PL |
| Pump | P |
| Switch, hand (knob) | HS |
| Switch, indicating, pressure (PIS-263-123D) | PIS |
| Transducer (calibrating unit) | TD |

| SYSTEMS | |
|---|---|
| Engineered safety features actuation system (RPS, PCIS, RBIS) | JE |
| Incore monitoring system (neutron monitoring system) | IG |
| Reactor Building | NG |
| Reactor recirculation system | AD |
| Residual heat removal system | BO |
| Solid state control system/Auxiliary logic control system (ATS) | JG |
| Standby gas treatment system | |