10 CFR 50.73

BOSTON EDISON
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Roy A. Anderson
Senior Vice President — Nuclear

September 21, 1992
BECo Ltr. 92-111

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) 92-011-00, "Unplanned Actuation of a Portion of the Residual Heat Removal System Logic Circuitry During Surveillance Testing," is submitted in accordance with 10 CFR Part 50.73.

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

R. A. Anderson

DWE/bal

Enclosure: LER 92-011-00

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
U.S. Nuclear Re[illegible]atory Commission
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Standard BECo LER Distribution

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Pilgrim Nuclear Power Station	05000 293	1 OF 7

TITLE (4) Unplanned Actuation of a Portion of the Residual Heat Removal System Logic Circuitry During Surveillance Testing

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	05000
08	21	92	92	011	00	09	21	92	N/A	05000

OPERATING MODE (9)	N
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Douglas W. Ellis - Senior Compliance Engineer	508-747-8160

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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 August 21, 1992, at 2040 hours, an unplanned actuation of a portion of the Residual Heat Removal (RHR) System circuitry occurred during a logic system functional test. The actuation resulted in responses that included an automatic trip of Recirculation System motor-generator set/pump 'A', start of the two RHR System Loop 'B' pumps, and start of Emergency Diesel Generator 'B'. A 24-hour Limiting Condition for Operation (LCO) was entered because the Recirculation System Loop 'A' was not in service. The circuitry was reset and the affected systems were restored to normal service. The Recirculation motor-generator set/pump was restarted on August 22, 1992, at 0025 hours, and the 24-hour LCO was terminated.

The root cause of the event was utility non-licensed technician error. The technician mistakenly actuated the wrong relay at a step in the test procedure. A performance assessment of the surveillance team and technician was conducted. Preventive action has been or is being taken and planned to preclude recurrence.

This event occurred during power operation with the reactor mode selector switch in the RUN position. The reactor power level was 100 percent. The Reactor Vessel (RV) pressure was 1028 psig with the RV water temperature at 550 degrees Fahrenheit. This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) and this event posed no threat to the public health and safety.

REQUIRED NUMBER OF DIGITS/CHARACTERS
FOR EA BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
3	VARIES	PAGE NUMBER
4	UP TO 76	TITLE
5	6 TOTAL 2 PER BLOCK	EVENT DATE
6	7 TOTAL 2 FOR YEAR 3 FOR SEQUENTIAL NUMBER 2 FOR REVISION NUMBER	LER NUMBER
7	6 TOTAL 2 PER BLOCK	REPORT DATE
8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Pilgrim Nuclear Power Station	05000 293	92	- 011 -	00	2 OF 7

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

Background

The High Pressure Coolant Injection (HPCI) System is designed to automatically initiate if a low-low Reactor Vessel (RV) water level condition or high Drywell pressure condition occurs. The system is also designed to automatically trip if a high RV water level condition occurs. The low-low RV water level condition is signaled to the HPCI System circuitry via the Residual Heat Removal (RHR) System Channel 'A' relays 10A-K7A/-K8A and/or Channel 'B' relays 10A-K7B/-K8B.

The HPCI System logic circuitry is functionally tested in accordance with procedures to meet Technical Specification 4.2.B/Table 4.2.B. Procedure 8.M.2-2.10.4-2, "HPCI Initiation Logic/High Water Level Trip Reset Test," functionally tests a portion of the logic circuitry. This test includes the use of a test switch that enables the individual testing of trip relays (e.g., 10A-K8B) and related HPCI System circuitry and components.

The RHR System Low Pressure Coolant Injection (LPCI) function is designed to automatically initiate if a low-low RV water level condition or high Drywell pressure condition occurs. The High Drywell pressure condition is signaled by Channel 'A' trip relays 10A-K5A/-K6A and/or Channel 'B' trip relays 10A-K5B/-K6B to the Channel 'A' logic relay 10A-K10A and Channel 'B' logic relay 10A-K10B.

Just prior to the event, plant systems status was:

- A portion of the HPCI System logic circuitry was being functionally tested in accordance with Procedure 8.M.2-2.10.4-2 (Rev. 17) Attachment 1.

- The Emergency Diesel Generators (EDGs) 'A' and 'B' were in standby service. The Auxiliary Power Distribution System was energized from the Main Generator via the Unit Auxiliary Transformer. The switchyard air type circuit breakers 102, 103, 104, and 105 were closed. The 23 KV distribution system was energized. The Shutdown Transformer was in standby service.

- The Recirculation System motor-generator (MG) set/pump 'A' and the MG set/pump 'B' were operating at approximately 95 percent speed. The total reactor core flow was approximately 66.5 Mlbs/hour.

- The RHR System was in standby service and configured for the LPCI function.

- The Reactor Core Isolation Cooling System was in standby service.

- The Feedwater Level Control System was in the three element control mode.

REQUIRED NUMBER OF DIGITS/CHARACTERS
FOR EACH BLOCK

BLOCK NUMBER	NUMBER OF DIGITS/CHARACTERS	TITLE
1	UP TO 46	FACILITY NAME
2	8 TOTAL 3 IN ADDITION TO 05000	DOCKET NUMBER
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8	UP TO 18 -- FACILITY NAME 8 TOTAL -- DOCKET NUMBER 3 IN ADDITION TO 05000	OTHER FACILITIES INVOLVED
9	1	OPERATING MODE
10	3	POWER LEVEL
11	1 CHECK BOX THAT APPLIES	REQUIREMENTS OF 10 CFR
12	UP TO 50 FOR NAME 14 FOR TELEPHONE	LICENSEE CONTACT
13	CAUSE VARIES 2 FOR SYSTEM 4 FOR COMPONENT 4 FOR MANUFACTURER NPRDS VARIES	EACH COMPONENT FAILURE
14	1 CHECK BOX THAT APPLIES	SUPPLEMENTAL REPORT EXPECTED
15	6 TOTAL 2 PER BLOCK	EXPECTED SUBMISSION DATE

- The reactor mode selector switch was in the RUN position with the reactor at 100 percent power. The RV was 1028 psig with the RV water temperature at approximately 550 degrees Fahrenheit. The RV water level was steady at approximately +30 inches.

EVENT DESCRIPTION

On August 21, 1992, at 2040 hours, an unplanned actuation of a portion of the RHR System logic circuitry occurred during a functional test of the HPCI System logic circuitry. The actuation resulted in responses that included:

- Automatic start of EDG 'B'. The EDG did not load onto the respective 4160 VAC bus because the bus remained energized from the Unit Auxiliary Transformer.

- Automatic selection of RHR System Loop 'A' for the LPCI function. The selection included the following responses:

    - The RHR System Loop 'A' injection valve MO-1001-28A remained in the fully open position. The in-series injection valve MO-1001-29A remained in the closed position because the RV pressure was greater than the setpoint (calibrated at approximately 375 psig decreasing) that permits the valve to open.

    - Automatic repositioning of the open RHR System Loop 'B' injection valve MO-1001-28B to the closed position. The in-series injection valve MO-1001-29B remained in the closed position. The RHR System Loop 'B' heat exchanger bypass valve MO-1001-16B opened.

    - Automatic closing of the Recirculation System Loop 'A' pump discharge valve MO-202-5A, and trip of the motor-generator (MG) set/pump 'A' due to the actuation of the closing coil of valve MO-202-5A. Reactor power decreased to approximately 70 percent because of the decrease in the total core flow due to the reduced recirculation flow.

- Automatic start of RHR System Loop 'B' pumps P-203B/D and flow to the Torus through the minimum flow piping and valve MO-1001-18B.

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

Licensed operator response to the event was orderly and included the following actions. The reactor core flow was calculated at approximately 37.5 Mlbs/hour and determined to be in the permitted region of the reactor power-flow operating map. The Average Power Range Monitors and Low Power Range Monitors were monitored for neutron flux instability oscillations with no oscillations evident. At 2048 hours, the RHR System pumps P-203B/D were returned to standby service. The RHR System circuitry was reset and the affected valves were returned to the standby configuration for the LPCI function. The HPCI System was returned to standby service. At 2056 hours, valve MO-202-5A was opened. By 2149 hours, reactor power had been reduced to approximately 46 percent by the sequential insertion of applicable control rods and adjustment of the Recirculation System Loop 'B' flow.

A 24-hour Limiting Condition for Operation (LCO) was entered in accordance with Facility Operating License condition 3.E because the Recirculation System Loop 'A' was not in service.

Problem Report 92.9149 was written to document the event. The NRC Operations Center was notified in accordance with 10 CFR 50.72 on August 21, 1992, at 2301 hours.

After verifying the cause of the actuation and systems' response, the Recirculation System MG set/pump 'A' was started on August 22, 1992, at 0025 hours, and the 24-hour LCO was terminated. Reactor power was subsequently increased via control rod withdrawal and increased recirculation/core flow. Full reactor power operation was achieved on August 22, 1992, at 0910 hours.

CAUSE

A critique of the event was conducted on August 21, 1992, and was attended by applicable personnel including the Instrumentation and Control (I&C) technicians who were performing the surveillance test.

The root cause of the event was utility non-licensed I&C Technician error. The technician mistakenly actuated RHR System Channel 'B' relay 10A-K10B instead of relay 10A-K8D at step [24] of Procedure 8.M.2-2.10.4-2, Attachment 1. These semi-flush mounted General Electric type HFA relays are located on panel C-933 approximately 18 inches apart and are approximately five and one-half feet above the floor level. The relays are individually identified with a nameplate (e.g., 10A-K8B). A factor contributing to the error was that Procedure 8.M.2-2.10.4-2, although including double verification for steps including but not limited to lifting/relanding electrical jumpers, did not include a double verification for the relay to be actuated.

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

The verification of the cause of the selection of the RHR System Loop 'A' for the LPCI function concluded Loop 'A' was selected because the Recirculation System jet pump riser differential pressure for Loop 'A' was greater than Loop 'B'. For this event, a Recirculation System pipe break did not exist; consequently, the Recirculation/RHR System Loop 'B' would have been expected to be selected for the LPCI function. Loop selection is initiated by differential pressure switches DPIS-261-12A/B/C/D. These switches sense differential pressure between Recirculation System Loop 'A' and 'B' jet pumps and function to initiate the associated trip relays if the jet pump riser differential pressure for Loop 'A' is greater than Loop 'B'. The pressure switches are functionally tested monthly and calibrated quarterly in accordance with Procedure 8.M.2-2.2.2, "Recirculation Jet Pump Riser Differential Pressure". The setpoint for the recirculation jet pump riser differential pressure trip function (i.e., pressure switches) is specified in Technical Specification Table 3.2.B at greater than 0.5 psid and less than 1.5 psid. The pressure switches were recently calibrated and functionally tested with satisfactory results in accordance with Procedure 8.M.2-2.2.2 (Rev. 14), Attachment 2, on June 16, 1992. The as-left setpoints of the pressure switches were 0.90 +/- 0.05 psid. The RHR System Loop 'A' was selected for the LPCI function because the jet pump riser differential pressure for Loop 'A' was greater than Loop 'B' at the completion of a designed pipe break detection time delay (approximately two seconds) that was initiated as a result of the manual actuation of relay 10A-K10B.

CORRECTIVE ACTION

A performance assessment of the surveillance team and responsible technician was conducted. As a result of the assessment, the responsible I&C Technician was counseled.

PREVENTIVE ACTION

This event was discussed with all I&C Supervisors on August 24, 1992, and was discussed during the weekly I&C Laboratory meeting on August 28, 1992.

A change to procedure 8.M.2-2.10.4-2 (Rev. 17) was initiated. The change will be reviewed/approved prior to the next performance of the surveillance and includes:

- The addition of a precaution at the beginning of the test to double verify the correct relay is identified prior to performing the step.

- The addition of a double verification to those procedure steps that require the manual actuation of a relay.

An assessment of other surveillance procedures was being conducted when this report was prepared and is currently expected to be completed by the end of December 1992. The assessment may identify other possible improvements to preclude recurrence.

This event will be included as part of the Plant Status Update training for I&C Technicians and Supervisors.

SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The actuation was the designed response to the conditions existing at the time relay 10A-K10B was manually actuated.

There were no component or system failures that caused or resulted from this event.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the actuation, although a designed response to the conditions existing at the time relay 10A-K10B was manually actuated, was not an expected or planned part of the activity being performed (Procedure 8.M.2-2.10.4-2).

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review focused on LERs involving a similar actuation due to personnel error. The review identified similar events reported in LERs 50-293/84-012-00 and 89-027-00.

For LER 84-012-00, an unplanned start of EDG 'B' occurred while shut down for refueling on September 20, 1984. The event occurred during the performance of Procedure 8.M.2-2.10.1-8, "Logic System Functional Test of the 'A' Core Spray System - High Drywell Pressure Auto Initiation Trip". The event occurred when an I&C Technician manually actuated relays 14A-K5B and 14A-K6B simultaneously instead of manually actuating only relay 14A-K5B. To preclude recurrence, personnel were cautioned to be careful while performing the surveillance test.

For LER 89-027-00, an unplanned actuation of a portion of the RHR System logic circuitry occurred while shut down on September 5, 1989, at 1805 hours. The event occurred during logic relay testing at step 2.6 of Procedure TP 88-78 (Rev. 3), "General Electric CR2820 Time Delay Relays for ADS, Core Spray, RHR, and RPS Systems", Attachment 2. The systems' responses included an automatic start of EDG 'A', closing of the Recirculation System Loop 'B' pump discharge valve MO-202-5B, opening of the RHR Loop 'B' injection valve MO-1001-28B, closing of the RHR Loop 'A' injection valve MO-1001-29A, and opening of the RHR Loop 'A' heat exchanger bypass valve MO-1001-16A. The cause of the event was I&C Technician error. The technician jumpered the termination points (1-2) of RHR System relay 10A-K10A instead of Core Spray System relay 14A-K10A. Corrective action taken included the revision of TP 88-78 to include verification of those procedural steps involving the installation of a jumper or insulating boot.

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

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Generator Set, Motor	MG
Relay (10A-K10B)	RLY
Valve (MO-202-5A)	V
Valve, control, temperature (MO-1001-16B)	TCV
Valve, injection (MO-1001-28A/B, -29A/B)	INV

SYSTEMS	
Emergency Onsite Power Supply System (EDG 'B')	EK
Engineered Safety Features Actuation System	JE
Reactor Recirculation System	AD
Residual Heat Removal System (RHR/LPCI)	BP