



BOSTON EDISON

Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

Ralph G. Bird
Senior Vice President — Nuclear

January 8, 1990
BECo Ltr 90-004

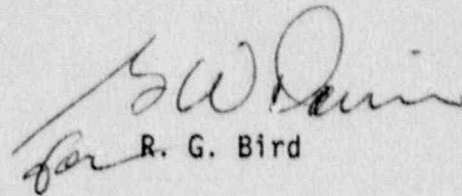
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

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

Dear Sir:

The enclosed Licensee Event Report (LER) 89-039-00, "Automatic Closing of the Primary Containment System Group 3 Isolation Valves While Shutdown", 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.


for R. G. Bird

DWE/bal

Enclosure: LER 89-039-00

cc: Mr. William Russell
Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Rd.
King of Prussia, PA 19406

Sr. NRC Resident Inspector - Pilgrim Station

Standard BECo LER Distribution

*cert No
P082895991*

JE22

11

9001170318 900108
PDR ADOCK 05000293
S PIC

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Pilgrim Nuclear Power Station	DOCKET NUMBER (2) 050000293	PAGE (3) 1 OF 5
--	--------------------------------	--------------------

TITLE (4)
Automatic Closing of the Primary Containment System Group 3 Isolation Valves While Shutdown

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 NAMES			DOCKET NUMBER(S)		
1	2	0	9	8	9	8	9	0	N/A			050000		
1	2	0	9	8	9	0	3	9	N/A			050000		

OPERATING MODE (9) N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)									
POWER LEVEL (10) 000	20.402(b)		20.406(e)	<input checked="" type="checkbox"/>	50.73(a)(2)(iv)		73.71(b)			
	20.406(a)(1)(i)		50.36(e)(1)		50.73(a)(2)(v)		73.71(e)			
	20.406(a)(1)(ii)		50.36(e)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)			
	20.406(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)					
	20.406(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)					
20.406(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(ix)						

LICENSEE CONTACT FOR THIS LER (12)

NAME Douglas W. Ellis - Senior Compliance Engineer	TELEPHONE NUMBER
	AREA CODE: 508 7147 1816 0

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NRC

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) NO

EXPECTED SUBMISSION DATE (15)

MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On December 9, 1989 at 1245 hours, an automatic actuation of the Residual Heat Removal System (RHRS) portion of the Primary Containment Isolation Control System (PCIS) occurred while shutdown. The actuation occurred when the RHRS was being started for the shutdown cooling (SDC) mode of operation in accordance with procedure. The actuation resulted in the automatic closing of the Primary Containment System Group 3 (three)/SDC suction piping isolation valves.

The direct cause for the actuation was a hydrodynamic transient that actuated the protective high pressure (122 psig) switches for the SDC suction piping. The root cause is believed to have been some unvented air in the SDC suction piping. The air was most likely introduced into a section of the piping, while shutdown in October 1989, as a result of a valve bonnet leak and/or isolation that was subsequently conducted for valve sealing. The related accessible piping was inspected with satisfactory results. The PCIS logic circuitry was reset and the RHRS was satisfactorily put into service in the SDC mode of operation at 1501 hours. Corrective actions being planned include reconfiguring the existing vents in the SDC suction piping for improved venting.

This event occurred while in hot shutdown with the reactor mode selector switch in the SHUTDOWN position. The reactor power level was zero percent. The Reactor Vessel (RV) pressure was 5 (five) psig, and the RV water temperature was 230 degrees Fahrenheit with minor decay heat. 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.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Pilgrim Nuclear Power Station	DOCKET NUMBER (2) 0 5 0 0 0 2 9 3 8 9	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		8 9	0 3 9	0 0	0 2	OF	0 5

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

EVENT DESCRIPTION

On December 9, 1989 at 1245 hours, an automatic actuation of the Residual Heat Removal System (RHRS) portion of the Primary Containment Isolation Control System (PCIS) occurred while shutdown. The actuation occurred eight seconds after starting the RHRS pump 'A' for the shutdown cooling (SDC) mode of operation. The actuation resulted in the following designed responses:

- The inboard and outboard Primary Containment System (PCS)/Group 3 (three) SDC suction isolation valves (MO-1001-50 and -47), in the open position, closed automatically.
- The RHRS pump 'A' tripped automatically because the SDC suction valves were not fully open (i.e., the valves were closing).

Initial utility Control Room licensed operator response was to close an RHRS injection valve (MO-1001-28A) and investigate the cause for the actuation. At approximately 1445 hours, the PCIS circuitry was reset and the SDC suction valves (MO-1001-50 and -47) were reopened. At 1501 hours, the RHRS pump 'A' was started and the RHRS was satisfactorily put into service in the SDC mode of operation.

Failure and Malfunction Report 89-476 was written to document the event. The NRC Operations Center was notified on December 9, 1989 at 1515 hours.

This event occurred while in hot shutdown with the reactor mode selector switch in the SHUTDOWN position. The Reactor Vessel (RV) pressure was 5 (five) psig with the RV water temperature at 230 degrees Fahrenheit. The reactor power level was zero percent.

BACKGROUND

Just prior to the event, steady state shutdown operating conditions existed. The Recirculation System pumps 'A' and 'B' were not in service. The Reactor Water Cleanup (RWCU) System was in service. The RHRS was not in service in any of its operating modes.

A reactor scram had occurred on December 8, 1989 at 0308 hours. While still shutdown as a result of the scram (LER 89-038-00), a management decision was made on December 9, 1989 to achieve cold shutdown in order to perform planned maintenance and testing. The core decay heat was minor. However, the RV water temperature was greater than 212 degrees Fahrenheit and the SDC mode of operation was to begin in order to achieve cold shutdown conditions.

The RHRS (Loop 'A') was subsequently configured for the SDC mode of operation in accordance with section 5.4 of procedure 2.2.19 (Rev. 32) Attachment 6 (six), "RHR Shutdown Cooling Operations". The procedural steps taken included flushing and venting the SDC suction piping downstream of the outboard isolation valve (MO-1001-47) and, similarly, the discharge piping up to the Loop 'A' injection valve MO-1001-29A that is downstream and in-series with valve MO-1001-28A. At 1236 hours, the Recirculation System Loop 'B' pump (P-201B), in service at the time, was shutdown in accordance with procedure 2.2.84, "Reactor Recirculation System".

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Pilgrim Nuclear Power Station	DOCKET NUMBER (2) 0 5 0 0 0 2 9 3	LER NUMBER (6)			PAGE (3)	
		YEAR 8 9	SEQUENTIAL NUMBER — 0 3 9	REVISION NUMBER — 0 0 0	3 OF 0 5	

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

Prior to commencing the SDC mode of operation, a pre-evolutionary briefing was conducted in accordance with procedure 1.3.37, "Conduct of Operations". At 1244 hours, the RHRS Loop 'A' (throttling type) injection valve MO-1001-28A was closed and the in-series injection valve MO-1001-29A was opened in accordance with procedure (2.2.19 Attachment 6 section 5.4) steps [10] and [11], respectively. The SDC suction valves (MO-1001-50 and -47) were then opened in accordance with procedure step [12]. The RHRS Loop 'A' pump 'A' (P-203A) was then started and valve MO-1001-28A was jogged open in accordance with procedure step [13] and a flow of approximately 3200 gpm was attained approximately five seconds later. The actuation occurred approximately eight seconds after the start of the pump.

CAUSE

A critique of the event was conducted on December 9, 1989 in accordance with procedure 1.3.63, "Conduct of Critiques and Incident Investigations". The critique was held to determine the cause for the event and was attended by appropriate personnel including the operators on shift at the time of the event.

The direct cause for the actuation was a hydrodynamic transient that actuated pressure switches (PS-261-23A and -23B) connected to the Recirculation System Loop 'A' pump suction piping. The SDC suction piping is connected to the Recirculation System Loop 'A' piping upstream of the Loop 'A' pump suction valve. The hydrodynamic transient probably resulted from some unvented air in the SDC suction piping. The setpoint for the pressure switches is calibrated at approximately 122 psig. The PCIS Group 3 logic circuitry is arranged as an interlock such that reaching the setpoint of either (or both) pressure switch results in a close signal to the related RHRS valves (MO-1001-50 and -47).

The root cause for the actuation is believed to have been some unvented air in the SDC suction piping. The air was most likely trapped in a section of the suction piping between the inboard isolation valve (MO-1001-50) and the outboard isolation valve (MO-1001-47). The air was most likely introduced into this section of piping as a result of a valve (MO-1001-50) bonnet leak and/or isolation that was conducted for subsequent leak sealing performed while shutdown in October 1989.

CORRECTIVE ACTION

The following corrective actions that have been taken include the following:

- Field inspection of accessible SDC suction piping located outside of primary containment (Drywell) was performed with satisfactory results. The inspections revealed no evidence of piping damage or unusual piping movement.
- The related instrumentation, alarms, logic circuitry, and valves (MO-1001-50 and -47) were functionally tested on December 11, 1989 with satisfactory results. The inboard valve related testing was performed in accordance with procedure 8.M.2-1.5.4 (Rev. 10) Attachment 'A', "RHR Isolation Valve Control - Test A - Inboard". The outboard valve related testing was performed in accordance with procedure 8.M.2-1.5.5 (Rev. 11) Attachment 'A', "RHR Isolation Valve Control - Test B - Outboard".

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Pilgrim Nuclear Power Station	DOCKET NUMBER (2) 0 5 0 0 0 2 9 3	LER NUMBER (6)			PAGE (3)	
		YEAR 8 9	SEQUENTIAL NUMBER — 0 3 9	REVISION NUMBER — 0 0	0 4	OF 0 5

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

Corrective Actions being planned include the following:

- The reconfiguration of existing vents that would more effectively eliminate trapped air in the SDC suction piping is part of the Long Term Plan (Item 31).

SAFETY CONSEQUENCES

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

The RHRS/SDC mode of operation has a power generation design basis only. The SDC mode of operation functions to reduce the RV water temperature to 125 degrees Fahrenheit approximately 20 hours after a shutdown for refueling or servicing activities.

If the actuation had occurred when the decay heat generation rate was greater, a gradual increase in the RV water temperature could have occurred. However, alternate means for heat removal are available and described in procedure 2.4.25, "Loss of Shutdown Cooling", including methods for feed and letdown using the Condensate System, RWCU System, and the Main Condenser.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the PCIS logic circuitry was actuated. The SDC mode of operation is not necessary to mitigate the consequences of an accident. The actuation was initiated by the 122 psig RV pressure switches (PS-261-23A and -23B). The pressure switches provide a protective function for the SDC suction piping. The SDC suction piping, extending from the Recirculation System Loop 'A' piping up to the RHRS pumps' suction valves MO-1001-43A/B/C/D, provides a flow path for the SDC mode of operation only.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(iv) involving the closing of the RHRS/SDC suction piping isolation valves. The review identified related events reported in LERs 50-293/87-008-01, 87-015-00, and 88-025-00.

For LER 87-008-01, an unplanned actuation of the RHRS portion of the PCIS occurred during refueling on October 15, 1987 at 2100 hours. The actuation resulted in the automatic closing of the isolation valves (MO-1001-50 and -47) and a temporary interruption in the SDC mode of operation. At the time of the event, the refueling cavity was flooded and the decay heat generation rate was negligible. The actuation occurred during a work task involving the installation of a new PCIS relay (16A-K69) in parallel with the inboard tripping relay (16A-K28) associated with PS-261-23A. The coil of relay 16A-K28 became de-energized during the installation of relay 16A-K69. The cause for the actuation was contractor electrical craft personnel error due to a deficient work plan.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Pilgrim Nuclear Power Station	DOCKET NUMBER (2) 0 5 0 0 0 2 9 3	LER NUMBER (6)			PAGE (3)	
		YEAR 8 9	SEQUENTIAL NUMBER 0 3 9	REVISION NUMBER 0 0		

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

For LER 87-015-00, actuations of the RHRS portion of the PCIS logic circuitry occurred while shutdown on December 7, 1987 at 1438 hours and on December 8, 1987 at 2145 hours. The December 7, 1987 actuation resulted in a temporary interruption in the SDC mode of operation. The actuation occurred while a utility I&C technician was installing a jumper to circuit contacts related to PS-261-23A in accordance with procedure 2.1.8.1 (Rev. 2), "Class 1 System Hydrostatic Test". During the jumper installation, a screw became disconnected and de-energized the coil of relay 16A-K28 that resulted in the event. The cause for the actuation was procedural inadequacy in that the procedure (2.1.8.1) did not adequately identify how the jumper was to be installed and did not contain a suitable caution regarding the potential impact to the SDC mode of operation. The December 8, 1987 actuation occurred at the beginning of the hydrostatic test when the RV pressure reached the setpoint of the pressure switches (PS-261-23A/B). The actuation resulted in the automatic closing of only the inboard isolation valve (MO-1001-50) because the outboard isolation valve (MO-1001-47) was in the closed position for the portion of the test (2.1.8.1 Rev. 2) being conducted. The cause for the actuation was procedural inadequacy in that the procedure (2.1.8.1) did not identify all the jumpers necessary to bypass the (122 psig) interlock for the test.

For LER 88-025-00, a low RV water level occurred while shutdown on December 3, 1988 at 0304 hours. The low RV water level that occurred (+3 inches narrow range level) resulted in actuations that included the RHRS portion of the PCIS logic circuitry. The isolation valves (MO-1001-50 and -47), in the open position, closed automatically. The low RV water level occurred following a local leak rate test of the SDC suction isolation valves when the outboard isolation valve (MO-1001-47) was opened (with valve MO-1001-50 in the open position). The valve was opened in accordance with procedure 2.2.86 (Rev.31), "Residual Heat Removal". The opening of valve MO-1001-47 resulted in the displacement of water from the RV (via the Recirculation System Loop 'A' pump suction piping) to an unfilled section of the 20 inch SDC suction piping. The cause for the actuation was utility licensed operator error. A contributing factor was a weakness in the procedure (2.2.86 Rev. 31) approved for the activity being performed.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS

Pump (P-203A)
Switch, Pressure (PS-261-23A/B)
Valve, Isolation (MO-1001-47 and -50)

CODES

P
PS
ISV

SYSTEMS

Engineered Safety Features Actuation System (PCIS)
Reactor Recirculation System
Reactor Services System (RHRS/SDC mode)

JE
AD
CF