NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Shoreham Nuclear Power Station Unit #1	0 5 0 0 0 3 2 2	1 OF 0 3

TITLE (4): Inadvertant RHR Loop "B" Trip in Shutdown Cooling Mode

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)
0 3	1 0	8 5	8 5	0 1 0	0 0	0 4	0 4	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (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 (AREA CODE)
Gary G. Rhoads, Operational Compliance Engineer	5 1 6 9 2 9 - 8 3 0 0

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)   [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On March 10, 1985 at 0350, an inadvertent isolation of the shutdown cooling system (Residual Heat Removal System) occurred while pressurizing the reactor vessel in preparation for backflushing the control rod drives to alleviate suspected plugging of the cooling water orifices. The plant was in Operational Condition 4, a condition not requiring the containment isolation system to be operational. Per an approved station procedure, the reactor vessel water level was established above the vessel flange and vessel heatup was performed utilizing the RHR system operating in the shutdown cooling mode . Subsequent to achieving a vessel temperature of 160 F to 180 F, reactor vessel pressurization commenced utilizing the control rod drive system. At approximately 88 to 90 psig, as indicated on the reactor vessel pressure indicator, the shutdown cooling system inboard isolation valve (E11*MOV047) automatically shut and RHR pumps E11*P014B and D tripped. The automatic isolation occurred as a result of exceeding the shutdown cooling system high suction pressure setpoint. Operations personnel manually shut the outboard shutdown cooling system isolation valve (E11*MOV048) and secured the shutdown cooling lineup. A maintenance work request was immediately issued to investigate the cause of the isolation and preparations were begun to start the reactor recirculation system to maintain the reactor vessel temperature.

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

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

On March 10, 1985 at 0350, an inadvertent isolation of the shutdown cooling system (Residual Heat Removal system) occurred while pressurizing the reactor vessel in preparation for backflushing the control rod drives to alleviate suspected plugging of the cooling water orifices. The plant was in Operational Condition 4, a condition not requiring the containment isolation system to be operational.

The control rod drive backflushing was being performed in accordance with an approved station procedure (TP 22.009.03). Per the procedure, the reactor vessel water level was established at approximately 24" above the vessel flange. Subsequent to achieving the desired water level, heatup was commenced utilizing the shutdown cooling mode (loop B) of the RHR system. Upon achieving a reactor vessel temperature of 160 F to 180 F, reactor vessel pressurization utilizing the control rod drive system commenced. At approximately 88 to 90 psig, as indicated on reactor vessel pressure indicator C32-PI003, the shutdown cooling system inboard isolation valve (E11*MOV047) automatically shut and both of the operating shutdown cooling pumps (E11*P014B and E11*P014D) automatically tripped. The automatic isolation occurred as a result of exceeding the shutdown cooling system high suction pressure setpoint. Operations personnel immediately shut the outboard shutdown cooling system isolation valve (E11*MOV048) and secured the shutdown cooling lineup. Additionally, a maintenance work request was issued to investigate the cause of the isolation and preparations were immediately begun to start the reactor recirculation system to maintain the reactor pressure vessel temperature at approximately 160 F to 180 F.

The automatic isolation of the inboard shutdown cooling isolation valve occurred due to exceeding the 119,+6,-6 psig setpoint associated with the shutdown cooling system suction pressure switch (B31*PS023A). Due to the pressure head associated with the flooded vessel (50 psig), a vessel steam dome pressure of 69 psig was sufficient to actuate the pressure switch (69 psig + 50 psig = 119 psig). Also, due to the location of the pressure indicator, only 20 psig of the water level pressure head was sensed by the indicator. This resulted in an indicated reactor pressure of 89 psig (69 psig + 20 psig = 89 psig). Thus, an indicated pressure of 89 psig corresponded to a steam dome pressure of 69 psig and a shutdown cooling system suction pressure of 119 psig.

The outboard isolation valve did not automatically shut because the trip setpoint of B31*PS023B was not achieved due to a slight difference between the setpoint of the A and B switches. Although both pressure switches were set within the specified range (119,+6,-6 psig), the B switch setpoint was 4 psig higher than the A switch. Thus, the outboard valve automatically shut but the inboard valve did not.

Although the control rod drive backflushing procedure contains a step directing the operators to secure the shutdown cooling system prior to exceeding 125 psig reactor pressure, the step does not alert the operator to the fact that the head of water in the vessel is sensed by the pressure switches but not observed by the pressure indicator.

To prevent recurrence, Operations personnel reviewed the incident in detail and are revising the following procedures to include a discussion of the relationship between reactor vessel pressure, reactor vessel water level pressure head, and shutdown cooling system suction pressure:

* SP 23.121.01 Residual Heat Removal (RHR) System

* TP 22.009.02 Control Rod Drive Backflushing at Elevated Reactor Pressure

All procedural changes will be completed by May 1, 1985.

LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION • P.O. BOX 628 • WADING RIVER, NEW YORK 11792
TEL. (516) 929-8300

April 4, 1985 PM 85-039

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

Dear Sir:

In accordance with 10CFR50.73, enclosed is a copy of Shoreham Nuclear Power Station Unit 1's License Event Report 85-010.

Sincerely yours,

W. Steiger

William E. Steiger, Jr.
Plant Manager

WES/gr

Enclosure

cc: Dr. Thomas E. Murley, Regional Administrator
Peter Eselgroth, Senior Resident Inspector
Institute of Nuclear Power Operations, Records Center
American Nuclear Insurers

SR.A21.0200

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