NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Perry Nuclear Power Plant Unit 1	0 5 0 0 0 4 4 0	1 OF 0 3

TITLE (4)

Test Instruction Deficiency and Personnel Errors Result In RHR System Isolations

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 5 0 0 0
0 7	0 5	8 6	8 6	0 3 2	0 0	0 8	0 1	8 6		0 5 0 0 0

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

OPERATING MODE (6)	4								
		20.402(b)	20.405(c)	X	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	0 0 0	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	
Paul Russ, Compliance Engineer, ext.6472	2 1 6	2 5 9 - 3 7 3 7

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 fifteen single-space typewritten lines) (16)

On July 5, 1986 at 0612 and July 12, at 1333 while in the shutdown cooling mode of operation, unexpected Residual Heat Removal (RHR) System isolations occurred. The isolations occurred during the performance of a monthly RHR surveillance test. For the first event, the Surveillance Test Instruction did not identify the necessary measures to prevent the isolation nor did it identify that performance of the test would cause an RHR isolation. During the second event, due to a lack of communications, plant technicians performed steps in the surveillance test out of sequence causing the second isolation. In both events, the RHR system responded as designed. In response to the isolations and RHR pump trip alarms, operators secured the performance of the surveillance test and verified proper plant conditions, reset the RHR isolation and restored the system to its previous configuration.

To prevent recurrence, the applicable RHR surveillance tests have been revised to include measures to prevent the RHR shutdown cooling isolation. Additionally, training will be conducted for personnel who review Surveillance Instructions regarding the identification and prevention of unnecessary system isolations. Plant technicians involved with the second event have been counseled regarding proper communications and strict compliance with instruction sequence. Retraining will be conducted for Instrument and Control technicians regarding instruction compliance and communication techniques.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

On July 5, 1986 at 0612 and July 12 at 1333 while in the Shutdown Cooling mode of operation, unexpected Residual Heat Removal (RHR)[BO] System isolations occurred. Prior to each event, the plant was in Operational Condition 4 (Cold Shutdown) with the RHR system loop A operating in the Shutdown Cooling mode and loop B in Standby Readiness. Reactor coolant temperature was approximately 150 degrees and reactor vessel [RPV] pressure atmospheric.

On July 5 at 0536, plant technicians commenced Surveillance Instruction (SVI)-B21-TO032A "Reactor Vessel Steam Dome Pressure and Reactor Vessel Pressure (RHR Cut-in Permissive) Channel Functional". At 0612, during the performance of step 5.1.16 a Shutdown Cooling RHR isolation occurred. The SVI did not provide the necessary measures to prevent the isolation nor did it identify that performance of the test would cause a Shutdown Cooling isolation. SVI-B21-TO032A had been performed successfully prior to this event, but the isolation had not been identified because the RHR system was shutdown and isolated at that time.

The RHR system responded to the isolation signal as designed causing an immediate actuation of the following isolation valves to their closed position; 1E12-FO08 Shutdown Cooling Outboard Suction Isolation [ISV], 1E12-FO23 RHR A Head Spray Isolation [ISV], 1E12-FO37A RHR A Upper Pool Cooling Isolation [ISV], 1E12-FO53A Shutdown Cooling A to Feedwater Shutoff [SHV]. In addition, RHR Pump [P] A automatically shutdown as designed when the 1E12-F008 suction isolation valve closed. Since this SVI affected only the Division 1 logic, only the outboard suction isolation valve and A loop components isolated. Division 2 provides redundant logic for the inboard suction isolation valve and B loop components.

In responding to the isolation and RHR pump trip alarms [ALM], operators secured performance of the SVI and verified reactor vessel pressure to be less than the isolation setpoint (135 psig). Operators then reset the RHR isolation and restarted the RHR pump with A loop lined up in the Shutdown Cooling Mode.

The SVI was recommenced on July 12, and included steps to prevent the shutdown cooling isolation. At 1333, during the performance of the revised steps, poor communications between two plant technicians in separate panel locations resulted in the instruction being performed out of sequence. A jumper lead, installed to prevent the isolation was removed prior to reconnecting a lifted wire to relay [RLY] 1B21H-K124A(P691), which resulted in the Shutdown Cooling isolation. The subsequent sequence of events was similar to that described above. On both occurrences, the period of time between the isolation and reinitiation of Shutdown Cooling was less than 16 minutes.

The reactor vessel pressure permissive for the RHR system (135 psig) is provided to protect the system for pressures exceeding its design. At pressures greater than 135 psig, shutdown cooling is accomplished via the main condenser [SG], or if unavailable, the steam condensing mode of RHR. If a similar isolation were to occur during a reactor shutdown and cooldown, with the maximum available

decay heat, the momentary loss of RHR cooling would not have had a significant effect on reactor coolant temperature. The operator actions taken to restore the RHR system to the Shutdown Cooling mode would have prevented the need for alternate shutdown cooling methods. Since Shutdown Cooling was restored to service in a timely manner and the fact that the system responded as expected, the event is not considered safety significant. One previous similar event occured on July 11, 1986 (see LER 86-034) when a technician neglected to perform all the necessary steps in a maintenance instruction and failed to jumper a relay in the RHR isolation logic. This resulted in an RHR isolation. Training for Instrument and Control technicians regarding the earlier event had not been completed prior to the events in this report.

To prevent recurrence, SVI's B21-TO032A through D have been revised to require the installation of an electrical jumper around the relay contact for the Shutdown Cooling isolation signal during the performance of the surveillance test. This jumper will prevent the closure of the isolation valves and the RHR pump trip. Additionally, training will be conducted for personnel who review Surveillance Instructions regarding the identification and prevention of unnecessary system isolations. Plant technicians directly involved with the second isolation have been counseled regarding proper communications when conducting plant evolutions and their responsibilities to maintain strict compliance with instruction sequence. Retraining will be conducted for Instrument and Control technicians regarding instruction compliance and communication techniques.

Energy Industry Identification System Codes are identified in the text as [XX].

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

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MURRAY R. EDELMAN
SR. VICE PRESIDENT
NUCLEAR

August 1, 1986
PY-CEI/NRR-0514 L

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

Perry Nuclear Power Plant
Docket No. 50-440
LER 86-032-0

Dear Sir:

Enclosed is Licensee Event Report 86-032-0 for the Perry Nuclear Power Plant.

Very truly yours,

A. Kaplan for

Murray R. Edelman
Senior Vice President
Nuclear Group

MRE:njc

Enclosure: LER 86-032-0

cc: Jay Silberg, Esq.
John Stefano (2)
K. Connaughton

U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL. 60137

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