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FACIL: 50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410
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RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-057-00: on 870921, ESF actuation occurred. Caused by cognitive personnel error & failure to follow procedure. Technician counseled. W/871020 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 5
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: 21 05000410

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	REG FILE 02	1	1	RES DEPY GI	1	1
	RES TELFORD, J	1	1	RES/DE/EIB	1	1
	RGN1 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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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)
Nine Mile Point Unit 2	0 5 0 0 0 410	1 OF 04

TITLE (4) Shutdown Cooling System Isolation due to Personnel Error; Failure to Follow Procedure

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)
											N/A	0 5 0 0 0
09	21	87	87	—	057	—	00	10	20	87	N/A	0 5 0 0 0

OPERATING MODE (9)	4	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)	20.405(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	
Robert G. Randall, Supervisor Technical Support	315	349-2445

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 September 21, 1987 at 1103 hours, Nine Mile Point Unit 2 (NMP2) experienced an Engineered Safety Feature (ESF) actuation, specifically, isolation of the Residual Heat Removal (RHS) Shutdown Cooling (SDC) system. At the time of the event, the plant was in the cold shutdown condition with the reactor mode switch in the "SHUTDOWN" position. Reactor pressure was atmospheric with a reactor coolant temperature of approximately 116°F.

The root cause of this event was cognitive personnel error; failure to follow procedure. The failure to follow procedure was caused by an inattention to detail.

Initial corrective actions were for the operators to identify the cause of the SDC isolation, verify the plant status as normal, and restore the SDC system to service.

Additional corrective actions for this event are:
1. The technician involved has been counseled.
2. A Training Modification Recommendation has been submitted requesting Instrument and Control (I&C) technician training on this event.
3. The event will be discussed in the I&C department safety meetings.

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

I. DESCRIPTION OF EVENT

On September 21, 1987 at 1103 hours, Nine Mile Point Unit 2 (NMP2) experienced an Engineered Safety Feature (ESF) actuation, specifically, isolation of the Residual Heat Removal (RHS) Shutdown Cooling (SDC) system. At the time of the event, the plant was in the cold shutdown condition with the reactor mode switch in the "SHUTDOWN" position. Reactor pressure was atmospheric with a reactor coolant temperature of approximately 116°F.

Niagara Mohawk Instrument and Control (I&C) technicians were in the process of performing the I&C surveillance procedure N2-ISP-LDS-M010, "Reactor Building General Area Temperature Instrument Channel Functional Test", when an RHS and Reactor Core Isolation Cooling (RCIC) isolation signal was received. This resulted in isolation of the the SDC system. No valve movement occurred for the RCIC system, since it was already isolated at the time of the event.

There were no components or systems which were inoperable and/or out of service which contributed to the event. No plant system or component failures resulted from the event.

II. CAUSE OF EVENT

The root cause of the event was cognitive personnel error; failure to follow procedure. This failure to follow procedure was due to an inattention to detail.

Procedure N2-ISP-LDS-M010 functionally tests the 10 reactor building general area temperature instrument channels 2RHS*TE81A and 2RHS*TE81B through 2RHS*TE85A and 2RHS*TE85B. The procedure is very repetitious, with essentially identical steps for the testing of each instrument channel. The bypass switch for the RHS and RCIC isolation logic is to be placed in "BYPASS" prior to performing each transmitter/switch calibration. Per procedure, the I&C technician is to request that the Chief Shift Operator (CSO) reposition the bypass switch to "BYPASS" for each instrument channel to be tested and verify the appropriate annunciator and computer point are activated. Preparing to test instrument channel 2RHS*TE82A, the lead I&C technician assumed he had ordered the bypass switch to the bypass position, without verifying the annunciator and computer point had alarmed, and instructed his co-worker to proceed with the procedure.

Subsequently, the lead I&C technician moved into position to verify the next step in the procedure. As the lead technician read over this step in preparation to verify its completion, he noted that the previous step to bypass the isolation logic was not initialed as having been performed. Once he had recognized his error, he attempted to inform his co-worker. However, by this time the co-worker was already in the process of performing the next step of the procedure. This step is to disconnect the thermocouple leads from the transmitter/switch input terminals. Without the trip logic bypassed, the SDC system isolation was initiated.

III. ANALYSIS OF EVENT

This event is considered reportable via 10CFR50.73 (a)(2)(iv) because the SDC isolation is an Engineered Safety Feature (ESF) function which is part of the Primary Containment and Reactor Vessel Isolation Control System (PCRVICS).

The SDC isolation did not impair the station's capability to achieve (or maintain) a safe shutdown condition, nor was there any conceivable impact to plant or public safety stemming from this event. This statement is based on the following: (1) The isolated system was quickly restored to service. (2) Additionally, even if we assume total failure of RHS shutdown cooling, other alternate methods for cooling the reactor are available and discussed in the NMP2 Final Safety Analysis Report Section 15.2.9..

As previously stated, the RCIC steam supply isolation valves also received an isolation signal, but no valve movement occurred since the valves were already closed. The RCIC system was not in operation (nor was it required to be operable by the NMP2 Technical Specifications Section 3.7.4) since the reactor was in a cold shutdown condition. Therefore, the RCIC isolation signal had no impact on station safety. However, at full power (or at any other condition where RCIC is required to be operable), if RCIC is not available to perform its intended function, the High Pressure Core Spray system would be available as a back up system.

The isolation function is considered to be a conservative ESF response. It is considered conservative since the primary objective of the isolation function is to provide protection to the plant and public by preventing releases of radioactive materials to the environment. Additionally, the SDC isolation function operated as designed with no other transients or inoperable systems contributing to this event.

The elapsed time for the event, from the isolation initiation to the restoration of SDC, was approximately three minutes.

IV. CORRECTIVE ACTIONS

Initial corrective actions were for the operators to identify the cause of the SDC isolation, verify the plant status as normal, reset the RHS/RCIC isolation logic and restore the SDC system to service.

Additional corrective actions include the following: (1) The I&C technician involved in this event has been counseled on the importance of following procedures when performing surveillance testing. (2) This event will be discussed in the regular I&C departmental safety meetings. This will assure that other I&C technicians are made aware of this event and its significance in a timely manner. (3) A Training Modification Recommendation (TMR# I87-23) has been initiated requesting discussion of this event in I&C technician training. Completion of this training is scheduled for February, 1988.

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

V. ADDITIONAL INFORMATION

Identification of Components Referred to in this LER

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
Isolation Valves	ISV	BO
Temperature Element	TT	IJ
Residual Heat Removal System	N/A	BO
Primary Containment	N/A	NH
Leakage Detection System	N/A	IJ
Primary Containment & Reactor Vessel Isolation Control System	N/A	JE
Reactor Core Isolation Cooling System	N/A	BN
High Pressure Core Spray System	N/A	BG
Bypass Switch	HS	IJ

There has been one previous related event which is discussed in LER 87-55. In this event, an I&C technician initiated an SDC system isolation by lifting the wrong lead while performing a similar type surveillance procedure. This event occurred five days prior to the event discussed in this report. Although some of the corrective actions for the event discussed in LER 87-55 are similar, the events are not considered similar. This is due to the significant contributing causes present in the previous event, namely human factors and design deficiencies.

NMP26474

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

301 PLAINFIELD ROAD
SYRACUSE, NY 13212

THOMAS E. LEMPGES
VICE PRESIDENT—NUCLEAR GENERATION

1987 OCT 26 A 8:53 USNRC-DS

October 20, 1987

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 87-57

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 87-57 Is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported."

A 10CFR50.72 report for this event was made at 1122 hours on September 21, 1987.

This report was completed in the format designated in NUREG-1022, Supplement No. 2, dated September 1985.

Very truly yours,

Thomas E. Lempges

Thomas E. Lempges
Vice President
Nuclear Generation

TEL/JTD/mjd

Attachments

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W. A. Cook

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