REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8709100195 DOC. DATE: 87/09/04 NOTARIZED: NO DOCKET #
FACIL: 50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410

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RECIP. NAME	RECIPIENT AFFILIATION

SUBJECT: LER 87-038-00: on 870807, RCIC primary containment valve isolation occurred. Caused by design deficiency. Work request issued to investigate event & track corrective actions. W/870904 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 7
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 05

TITLE (4) Reactor Core Isolation Cooling Steam Supply Isolation Valve Closed due to Design Deficiency

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)
08	07	87	87	038	00	09	04	87	N/A	0 5 0 0 0
									N/A	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 (9): 1

POWER LEVEL (10): 017

[ ] 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	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 August 7, 1987 at 1450 with the reactor at 17% power and the mode switch in "RUN", Nine Mile Point Unit 2 (NMP2) experienced an Engineered Safety Feature (ESF) actuation. This event consisted of a Reactor Core Isolation Cooling (RCIC) primary containment valve isolation. RCIC steam supply outboard isolation valve 2ICS*MOV121 closed due to a false RCIC high steam flow/instrument line break signal. The most probable cause of the signal was pressure oscillations in the RCIC steam supply line.

The root cause of the event has been determined to be a design deficiency. The setpoint of the trip unit producing the isolation has been evaluated as being too restrictive allowing pressure oscillations in the steam supply to isolate the RCIC turbine.

Immediate corrective action was to walk down the RCIC's steam line to identify any leaks. None were found. A Work Request (WR 123778) was then issued to investigate the event and track corrective actions. An Engineering and Design Coordination Report (E&DCR C95102) was later submitted to revise the isolation trip unit setpoints.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	PAGE (3)		
Nine Mile Point Unit 2	0 5 0 0 0 410	87	—	038	—	00	02	OF	05

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

I. DESCRIPTION OF EVENT

On August 7, 1987 at 1450 with the reactor at 17% power and the mode switch in "RUN", Nine Mile Point Unit 2 (NMP2) experienced an Engineered Safety Feature (ESF) actuation. This event consisted of a Reactor Core Isolation Cooling (RCIC) primary containment valve isolation (ICS*MOV121). At the time of the event station personnel were testing the steam condensing mode of the B Loop of the Residual Heat Removal (RHR) system. The steam supply line at ICS*MOV121 is common to both the RHR steam condensing system and the RCIC system. (See attachment for piping/instrument arrangement). The test was being performed per plant startup procedure N2-SUT-71-1, "Residual Heat Removal System".

The ESF actuation consisted of RCIC steam supply outboard valve 2ICS*MOV121 closing. Annunciator #323 on control room Panel 601, "Div I RCIC Steam Line Diff Press High" alarmed at this time. Immediate action was to walkdown the RCIC steam supply line for leaks. None were found. The Division I RCIC isolation signal was then reset, the steam line warmed and RCIC supply valve 2ICS*MOV121 reopened.

The steam supply line to the RCIC turbine is monitored for line break and for an instrument line break by two differential pressure transmitters (2ICS*PDT5A Div I, PDT5B Div II) located across two different elbows in the RCIC steam supply piping. Each differential transmitter supplies a signal to two trip units arranged in parallel. One unit will trip on high differential pressure indicating high flow (line break) or a break in the "low" side instrument line. The other unit will trip on a negative differential pressure indicating a break in the "high" side instrument line. A trip of either unit will result in applicable valve isolations and alarm control room annunciators.

Alarm #323 PNL601, "Div I RCIC Steam Line Diff Press High" receives its input from 2ICS*PDT5A. Associated trip units are E31-N683A, (high differential pressure) and E31-N690A (negative differential pressure). Initial evaluation indicated E31-N683A had tripped and caused the subsequent RCIC valve isolation. The trip was considered spurious since there was no steam flow past the flow elbow sensing lines for the affected trip unit (see attachment).

A Work Request (WR 123778) was issued on August 7, 1987 (later that afternoon) to troubleshoot trip unit E31-N683A. Surveillance procedure N2-ISP-ICS-M001, "Monthly Functional Test of RCIC Steam Line Flow High Instrument Channels" was performed as a functional check of E31-N683A. Per N2-ISP-ICS-M001, E31-N683A was found to perform satisfactorily and its setpoints found to be within specifications.

Upon further investigation, it was determined that trip unit E31-N690A most likely caused the isolation. The setpoint of E31-N683A and E31-N690A are +184.5 inches water and -20 inches water respectively. Pressure oscillations in the steam line would therefore most likely affect E31-N690A. An Engineering and Design Coordination Report was then issued to revise the setpoints of E31-N690A.

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

II. CAUSE OF EVENT

The root cause of the event has been determined to be a design deficiency. Trip units E31-N690A/N690B were originally set to trip and provide a RCIC steam valve isolation at a differential pressure of -20 inches water. A setpoint of -20 inches water has been evaluated as being too restrictive in that it allows pressure oscillations in the steam supply piping to isolate the RCIC turbine.

NOTE - Residual Heat Removal (RHR) valve 2RHS-PV21B provides pressure control of steam entering the RHR heat exchanger. On August 9, 1987, two days after the event, 2RHS*PV21B was found to be sticking due to its packing being too tight. Erratic action of this valve could have induced pressure oscillations (see attachment).

III. ANALYSIS OF EVENT

No adverse safety consequences occurred as a result of this event. With ICS*MOV121 closed, steam would be prohibited from reaching the RCIC turbine. The RCIC system would therefore have been inoperable until operations personnel reset the isolation signal, warmed the supply line and opened ICS*MOV121.

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

IV. CORRECTIVE ACTION

Immediate action was to walk down the RCIC steam supply line to look for leaks. No leaks were observed by Operations personnel. Initial evaluation indicated E31-N683A received a false signal and had caused a spurious trip. Work Request (WR 123778) was issued to troubleshoot trip unit E31-N683A. The trip unit was found to be functioning satisfactorily and its setpoints within specification per N2-ISP-ICS-M001, "Monthly Functional Test of RCIC Steam Line Flow High Instrument Channels". Further investigation showed the trip signal most likely originated from trip unit E31-N690A. The setpoint of E31-N690A/N690B has been revised from -20 inches water to -275 inches water per Engineering and Design Coordination Report (E&DCR) C95102. This change will allow pressure oscillations to occur and prevent a recurrence of a similar event. Trip unit E31-N683A is set at +184.5 inches water as required by Technical Specifications and will not be changed.

Work Request (WR 123763) was issued to troubleshoot 2RHS*PV21B and coordinate corrective actions. Per WR 123763 maintenance has adjusted the valve packing.

V. ADDITIONAL INFORMATION

Identification of Components Referred to in this LER

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
RCIC Isolation Valve (ICS-MOV121)	ISV	BN
Differential Pressure Transmitter	PDI	BN

No previous similar events have occurred at Nine Mile Point Unit 2.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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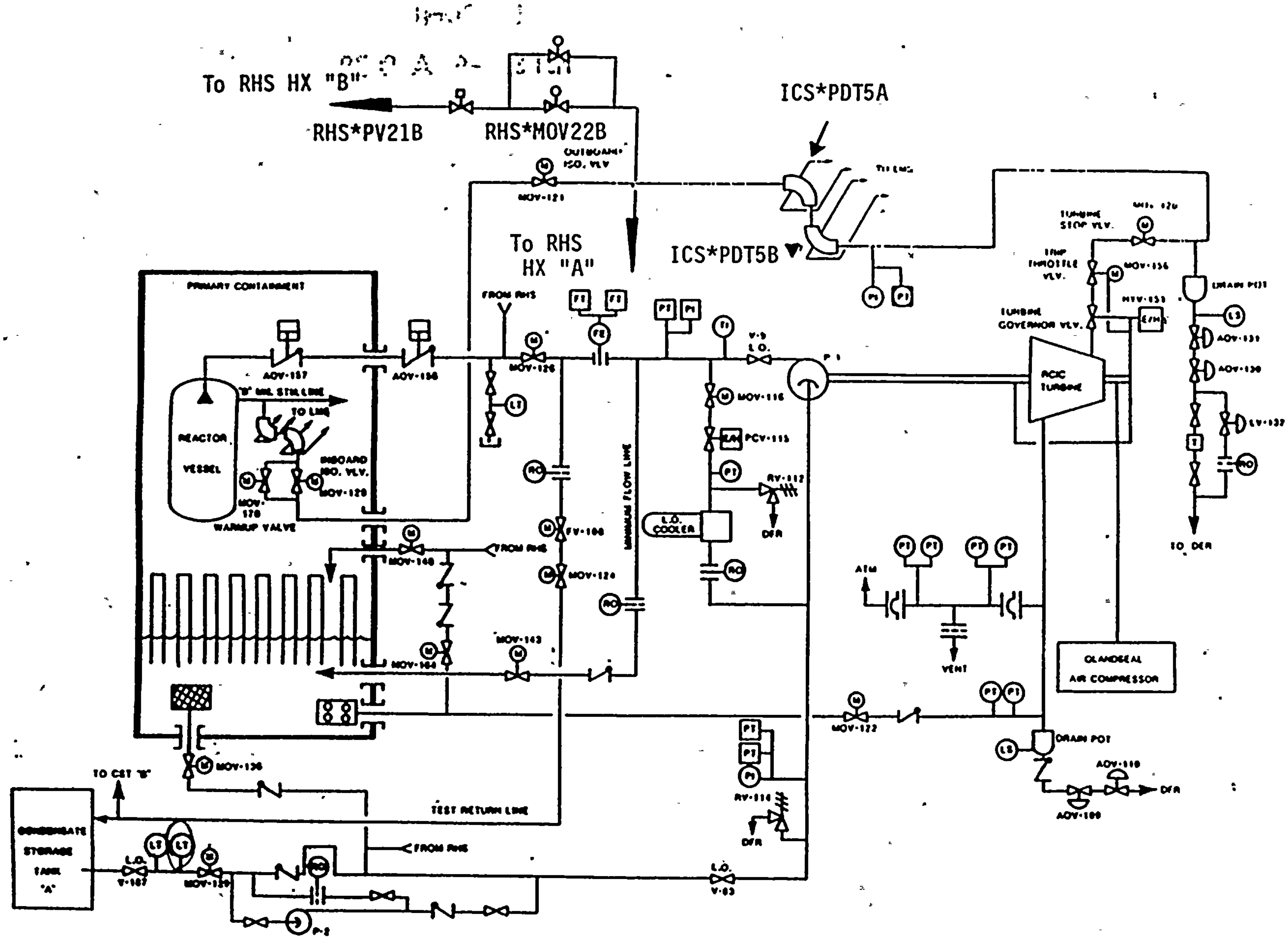
Nine Mile Point Unit 2	0 5 0 0 0 4 1 0	87	— 038 —	00	05 OF 05

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



NMP26445

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

301 PLAINFIELD ROAD
SYRACUSE, NY 13212

THOMAS E. LEMPGES
VICE PRESIDENT—NUCLEAR GENERATION

September 4, 1987

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

RE: Docket No. 50-410
LER 87-38

Gentlemen:

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

LER 87-38 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 10 CFR 50.72 report was made at 1542 on August 7, 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/JMT/mjd

Attachments

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

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