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FACIL: 50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410

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SUBJECT: LER 87-081-00: on 871226, reactor scrammed due to loss of condenser vacuum. Caused by failure of normal & high level control valves of fifth point feedwater heater. Cracked welds & level control valves to be repaired. W/880125 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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	NRR/DREP/RAB	1	1	NRR/DREP/RPB	2	2
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NRC Form 366

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Nine Mile Point Unit 2	0 5 0 0 0 410	1 OF 06

TITLE (4) Reactor Scram due to Loss of Condenser Vacuum Caused by Equipment Failure

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
12	26	87	87	— 081 —	00	01	25	88	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 026

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	
Robert E. Jenkins, Assistant Supervisor Technical Support	315	349-4220

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 December 26, 1987 at 1218 hours, Nine Mile Point Unit 2 experienced a reactor scram from approximately 26% of rated thermal power as a result of a turbine trip caused by low condenser vacuum. At approximately 0800 hours Control Room personnel noted that condenser vacuum was decreasing and that off gas system flow rates were increasing. Actions taken by operators to identify and stop the source of inleakage to the condenser were unsuccessful and the reactor subsequently scrammed at 1218 hours on the turbine trip.

The root cause of this event was malfunction of the normal (2HDL-FV5A) and high (2HDL-FV25A) level control valves of the fifth point feedwater heater (2HDL-E5A). Water level in 2HDL-E5A was fluctuating from high-high to low level which resulted in a large volume of water being periodically discharged to the condenser. These hydraulic transients caused the failure of condenser nozzle welds which allowed air inleakage to the condenser.

Immediate corrective actions were to reset the scram at 1225 hours, to follow the scram recovery procedure for safe shutdown of the plant, and to continue to investigate for the cause of the loss of vacuum. Work Requests have been completed to repair the cracked welds and the level control valves.

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1/1

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Nine Mile Point Unit 2	0 5 0 0 0 410	87	— 081	— 00	02 OF 06

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

I. DESCRIPTION OF EVENT

On December 26, 1987 at 1218 hours, Nine Mile Point Unit 2 (NMP2) experienced a reactor scram from approximately 26% of rated thermal power as a result of a turbine trip caused by low condenser vacuum.

At approximately 0800 hours, Control Room personnel noted that the condenser vacuum suddenly started to decrease. About that same time the Chief Shift Operator reported that the Offgas System (OFG) flow rate had increased and the Offgas Preheater inlet pressure had increased from 15 to 17 pounds per square inch absolute (psia). The condenser hotwell temperature was noted as 104 degrees Fahrenheit (°F). The offgas flow rate continued to gradually increase until fluctuations in flow rate developed. The flow rate swings occurred at one to one and one-half minute intervals and would swing from upscale to 10 to 15 standard cubic feet per minute (scfm) over a period of 10 to 20 seconds. Operations personnel were instructed to investigate the offgas system for signs of vacuum leakage.

Between 0830 and 0930 hours various level control problems were noted for Reheater Drain Tanks 2HDL-TK6A and TK6B. Feedwater heater 2HDL-E5A also had level control problems and was being controlled manually by use of its high level drain control valve 2HDL-LV25A. Condenser vacuum was noted as gradually decreasing during this time.

Between 1000 and 1030 hours the condenser hotwell temperature was noted as approximately 126°F with condenser vacuum degrading. Operations continued to search for the cause of the decreasing condenser vacuum.

Between 1030 and 1130 hours Operations personnel discovered that the sectionalizing gates in the cooling tower were not full open. Operators opened all gates, except one which appeared to have failed.

Reactor power had been systematically reduced since the start of the event in an effort to slow the loss of condenser vacuum. However, during the power reduction the Rod Sequence Control System (RSCS) blocked further rod movement before the Low Power Setpoint (LPSP) of 20% reactor power was reached. Power had been reduced to 26% and the rod pattern was being evaluated when the RSCS blocked further rod movement.

Before the rod pattern could be analyzed and rearranged to satisfy RSCS requirements the condenser vacuum reached its high vacuum setpoint of 22.1 inches of Mercury absolute. This resulted in a closure of the turbine stop valves and subsequent reactor scram.

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

Following closure of the turbine stop valves, reactor water level decreased to 155 inches which actuated Level 3 signals (low reactor water level, setpoint 159 inches). Closure of the turbine stop valves results in an increase in system pressure. This increased pressure collapses the voids in the reactor coolant, resulting in a lower water level for the same inventory of coolant in the vessel. As the turbine bypass valves open to control pressure, the reactor coolant swells as pressure is reduced. To prevent overfilling the vessel following closure of the turbine stop valves, a level setpoint setdown circuit is provided. This circuit reduces the low water level setpoint to provide a safe margin to avoid overfilling the vessel as the reactor coolant swells. The level setpoint setdown is engaged following a low water level signal maintained for 15 seconds. During this event the setpoint setdown circuit did not actuate however, because level was not below Level 3 for the required 15 seconds. Therefore, as the turbine bypass valves opened reactor water level rose to 206 inches, actuating all Level 8 signals (high reactor water level, setpoint 202.3 inches). The Level 8 signal caused a trip of Feedwater Pump A, as designed, to prevent reactor vessel overfill.

Immediate corrective actions were to reset the scram at 1225 hours, to follow the scram recovery procedure for safe shutdown of the plant, and to continue the investigation of the cause of the loss of condenser vacuum. The unit reached the cold shutdown condition by 0330 hours on December 28, 1987.

II. CAUSE OF EVENT

Investigation into the event determined that the immediate cause for the loss of condenser vacuum was air inleakage through two cracked welds at condenser nozzle 13A. This nozzle serves as a high level drain for the fifth point feedwater heater 2HDL-E5A. The condenser nozzle is an 18 inch diameter concentric reducer which is welded to a 16 inch pipe which runs to 2HDL-E5A. Cracks were noted at the weld which connects the reducer to the condenser and the weld which connects the reducer to the 16 inch pipe. The cracks were in the top of the configuration from approximately the ten o'clock position to the two o'clock position. The Engineering Department review of the failure of condenser nozzle 13A determined that a transient had caused forces which exceeded the allowable stresses of the condenser nozzle materials.

Further investigation showed that abnormal hydraulic transients had probably occurred as a result of 2HDL-E5A level control problems encountered during the event. On December 23, 1987 it was noted that the level transmitter (2HDL-LT5A) to normal level control valve 2HDL-LV5A was indicating zero output (zero level) which resulted in the valve closing and remaining closed. 2HDL-E5A level had to be controlled by its high level drain control valve 2HDL-LV25A. Water level was maintained by automatic modulation of 2HDL-LV25A. Normally, water level is maintained by automatic modulation of 2HDL-LV5A, which regulates drain flow to the fourth point feedwater heater 2HDL-E4A. When the normal drain is insufficient and the water level rises above a preset value, the high water level value will start modulating drain valve 2HDL-LV25A which bypasses to the condenser through condenser nozzle 13A. On December 27, 1987 it was found that 2HDL-LV25A had not been controlling level properly and water levels in 2HDL-E5A were fluctuating from high to low level. Thus, a large volume of water was being periodically discharged to the condenser through 2HDL-FV25A and condenser nozzle 13A.

The design calculations for condenser nozzle 13A did not predict high stresses at the nozzle. It has been postulated that the hydraulic or thermal transients encountered when water from 2HDL-E5A was periodically discharged through the nozzle caused the failure of the nozzle welds. The exact failure mechanism is still being investigated.

The cause of the failure of 2HDL-LT5A was a bent control linkage in level transmitter 2HDL-LT5A. The root cause of the event was the failure of 2HDL-LV25A to control level in 2HDL-E5A by modulating as designed.

III. ANALYSIS OF EVENT

This event is considered reportable via 10CFR50.73 (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)."

This event posed no adverse safety consequences. The turbine trip system and RPS functioned as designed. A low vacuum in the condenser trips the turbine to protect against high turbine exhaust pressure which could damage the turbine. The reactor scram is an analyzed event having been addressed in the NMP2 Final Safety Analysis Report (FSAR) Section 7.2 "Reactor Protection (trip) System (RPS) Instrumentation and Controls".

The RSCS is designed to reduce the consequences of a postulated rod drop accident to an acceptable level by constraining control rod movement to predetermined patterns and sequences. The RSCS is required to be operable when reactor power is below the LPSP of 20%. Thus, the RSCS rod blocks which occurred during the event was a conservative action designed to maintain the reactor core in an analyzed condition and posed no safety hazard. The operators were in the process of changing the rod pattern to satisfy RSCS requirements at a power level of 26% when the RSCS rod blocks occurred. Thus, the system acted more conservatively than designed.

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The total duration from the start of the loss of condenser vacuum until the reactor scram was approximately four and one quarter hours. The total duration of the event from reactor scram to resetting the scram signal was approximately seven minutes.

IV. CORRECTIVE ACTIONS

Immediate corrective actions were to reset the scram at 1225 hours, to follow the scram recovery procedure to shutdown the plant, and to continue the investigation to determine the cause for the loss of condenser vacuum. Further corrective actions are as follows:

1. A Work Request (WR 130708) was written to repair the cracked condenser nozzle welds. The welds were repaired and inspected by December 29, 1987.

2. The Engineering Department performed a review of the failure of condenser nozzle 13A. They found the as built configuration was consistent with the design calculations, which did not predict high stresses at the nozzle. They found that parallel pipes of identical service had not exhibited similar signs of stress or fatigue. It was recommended that this line should be returned to service.

3. The Engineering Department requested that a walkdown of the feedwater heater drain system piping be performed during a heatup and/or at the time high water level cycling in 2HDL-E5A is encountered. These walkdowns were performed by power ascension piping engineers during the subsequent plant startup.

4. A Work Request (WR129387) was issued to repair the high level control valve 2HDL-LV25A. Work on this WR was completed on December 30, 1987. However, Administratively the valve is not considered operable until the paperwork is complete.

5. A Work Request (WR 129369) was issued to repair the level transmitter to 2HDL-LV5A which was causing the valve to remain closed. The cause of the failure was found to be a bent control linkage. The linkage was straightened and 2HDL-LV5A was returned to service on December 26, 1987.

6. The rod withdrawl sequence has been revised. A change to the LPSP will be processed after more operational data has been collected if such a change is warranted.

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V. ADDITIONAL INFORMATION

There has been one previous reactor scram at NMP2 caused by a loss of condenser vacuum (LER 87-64). That event was caused by personnel error however, and not by equipment malfunction as was the case in this event. Thus, the events are not considered to be similar.

The condenser nozzle which failed is an 18 inch diameter concentric reducer which is welded to a 16 inch pipe. The piping is ANSI Class 302 designed and constructed according to ANSI B 31.1.0, "Power Piping".

Level Transmitter 2HDL-LT5A is a 12000 series liquid level controller manufactured by Masoneilan International, Inc., Model No. 12829.

Identification of Components Referred to in this LER

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
Turbine	TRB	N/A
Condenser	COND	N/A
Reactor	RCT	N/A
Reactor Protection System (RPS)	N/A	JC
Feedwater System (FWS)	N/A	SJ
Offgas System (OFG)	N/A	SH
Turbine Trip System (ETS)	N/A	JJ
Feedwater Pump Trip System	N/A	JK
Rod Sequence Control System (RSCS)	N/A	JD
Feedwater Heater	HX	SM
Level Control Valve	LCV	SM
Nozzle	NZL	SM
Level Transmitter	LIT	SM

NMP29849

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

301 PLAINFIELD ROAD
SYRACUSE, NY 13212

THOMAS E. LEMPGES
VICE PRESIDENT—NUCLEAR GENERATION

January 25, 1988

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

RE: Docket No. 50-410
LER 87-81

Gentlemen:

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

LER 87-81 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 (b) (2) (ii) report was made at 1218 hours on December 26, 1987.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

Thomas E. Lempges

Thomas E. Lempges
Vice President
Nuclear Generation

TEL/CDS/mjd

Attachments

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

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