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FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
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FIRLIT,J.F. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 92-001-00:on 920122,determined that min number of operable instrument channels per operable trip sys required by TS not met due to personnel error.Reactor power level raised to clear condition.W/920302 ltr.

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NIAGARA MOHAWK

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Joseph F. Firlit
Vice President
Nuclear Generation

NMP84829

March 2, 1992

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

RE: Docket No. 50-220
LER 92-01

Gentlemen:

In accordance with 10CFR50.73, we hereby submit the following Licensee Event Report:

LER 92-01 Which is being submitted in accordance with 10CFR50.73 (a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications."

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

A one week extension (to 2/28/92) was granted for this LER by the NRC Senior Resident Inspector.

Very truly yours,

FOR JOSEPH F. FIRLIT
V.P. NUCLEAR GENERATION

Joseph F. Firlit
Vice President - Nuclear Generation

JFF/AC/lmc
ATTACHMENT

xc: Thomas T. Martin, Regional Administrator Region I
Wayne L. Schmidt, Senior Resident Inspector

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1/1

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)
Nine Mile Point Unit 1	0 5 0 0 0 2 2 0	1 OF 0 7

TITLE (4) Violation of Technical Specifications Due to Lack of Configuration Management Caused by Personnel Error

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
0 1	2 2	9 2	9 2	0 0 1	0 0	0 3	0 2	9 2	N/A	0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 7 3	20.402(b)	20.405(c)	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)	X 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
R. L. Tessier, Manager Operations NMP1	3 1 5 3 4 9 - 2 7 0 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 January 10, 1992, at 2309 hours, with the mode switch in "RUN" and reactor power level at 72 percent of rated, Nine Mile Point Unit One (NMP1) was in a condition prohibited by plant Technical Specifications (TS). The condition existed for five hours and fifteen minutes without plant personnel being aware of the violation.

On January 22, 1992, at 0700 hours, with the mode switch in "RUN" and reactor power level at approximately 73 percent of rated, NMP1 was in a condition prohibited by plant TS. It was determined that the minimum number of operable instrument channels per operable trip system required by plant Technical Specifications were not met.

The immediate cause for both events was an instrumentation root valve for the Reactor Protection System found closed, combined with a leaking instrument drain valve. The root cause was lack of configuration management, i.e., the root valve was not identified on station prints or valve lineup procedures.

The immediate corrective action was to raise reactor power level to clear the condition. Additional corrective actions included: issue a Deviation Event Report to evaluate the event, implement a Temporary Modification to maintain the required number of operable instrument channels per operable trip system, and issue a work request to investigate the instrumentation.

Long term corrective actions include reviewing the Root Valve Verification program for other RPS input instrumentation, revising station prints and procedures, repair/replacement of the leaking instrument drain valve, and train personnel.

I. DESCRIPTION OF EVENT

Description of 1/11/92 Event

On January 10, 1992, at 2309 hours, with the mode switch in "RUN" and reactor power level at 72 percent of rated, NMP1 was in a condition prohibited by plant Technical Specifications. Specifically, the turbine stop valve position and the generator load rejection scrams were indicating bypassed at greater than 45 percent reactor power. The cause of this condition was not identified until January 22, 1992.

NMP1 began to reduce power at 2145 hours, 1/10/92, for a rod sequence exchange. Shortly after reaching 72 percent reactor power (2307 hours) the following annunciator alarm was received "TURBINE FIRST STAGE BOWL PRESSURE LOW". Process Computer printout identified the following times and computer points in alarm condition, 2309 hours - F046, "1ST STG BOWL PR 12 BYP" and 2340 hours - F045, "1ST STG BOWL PR 11 BYP." The alarm was cleared when power was increased at the conclusion of the rod sequence exchange.

Operations personnel discussed the condition of the alarm and recognized that it was not expected for present plant conditions. Work Request #201174 was generated 1/11/92 stating the alarm was in at 65 percent power and should not alarm until 45 percent power and requested a calibration.

The work request was dispositioned the following Monday 1/13/92 as a priority 1A (Emergency Work). The cause of failure on WR# 201174 indicated switch 02-13A was slightly low and should not have caused the annunciator to alarm. The work request was accepted and closed 1/14/92. The condition of the drain valve and the root valve remained undiscovered until 1/22/92.

A Deviation Event Report was not generated and therefore the reportability of the alarm was not determined.

Description of 1/22/92 Event

On January 22, 1992, at 0700 hours, with the mode switch in "RUN" and reactor power level at approximately 73 percent of rated, Nine Mile Point Unit 1 (NMP1) was in a condition prohibited by plant Technical Specifications. Specifically, it was determined that there were less than the minimum number of operable instrument channels per operable trip system as required by Technical Specification Table 3.6.2.a (10) and (11). Technical Specification Section 3.6.2.a (1) requires the rods to be inserted for this condition.

I. DESCRIPTION OF EVENT (cont.)

At the time of the event, NMP1 was reducing power in order to reverse Circulating Water System flow to remove ice buildup at the circulating water intake structure. During the power reduction, annunciator F3-4-6, "Turbine First Stage Bowl Pressure," alarmed. This alarm indicates that the turbine stop valve closure and generator load rejection scram functions are bypassed. These scram functions may be bypassed only when reactor power is less than 45 percent. At the time of the event, the reactor power was greater than 45 percent with turbine first stage bowl pressure approximately 630 pounds per square inch gauge (psig). The alarm setpoint corresponding to 45 percent thermal power is calibrated at 310 psig. Operators were not able to verify the alarming condition in accordance with annunciator response procedure due to the process computer being taken out of service for maintenance at 0443 hours. After plant conditions requiring reverse flow stabilized, reactor power was increased to clear the alarming condition.

Discussions of the alarm condition resulted in the Operations Branch initiating a Deviation Event Report (DER) at 0930 hours. A Temporary Modification was implemented that ensured the turbine stop valve closure and generator load rejection scrams would be engaged over the entire range of reactor power. Defeating the ability to bypass these scrams is more conservative than that required by Technical Specifications.

A Work Request (WR# 201202) was initiated to investigate the problem. Test pressure gauges were connected to both instrument channel sensing lines. Sensing line to pressure switches 02-13C and 02-13D was at correct pressure. Sensing line to pressure switches 02-13A and 02-13B read approximately one-half of known turbine first stage bowl pressure. The leaking instrument drain valve was opened and reseated and the line remained depressurized. Operations walked the sensing line down and discovered the root valve closed. The root valve was opened and sensing pressure at pressure switches 02-13A and 02-13B returned to normal.

On January 23, 1992, reactor power was again reduced to return Circulating Water System flow to normal line up. During this power reduction, the turbine first stage bowl pressure annunciator alarm did not activate, which indicated that with the root valve fully open and the instrument drain valve reseated, the instrumentation was functioning properly. The temporary modification which was installed was removed, returning the logic to normal configuration.

II. CAUSE OF EVENT

The immediate cause of the event was the misalignment of the root valve in the sensing line to pressure switches 02-13A and 02-13B, combined with the leaking instrument drain valve which permitted sensing line pressure to decrease to below the instrument setpoint. Apparently, there was enough seat leakage in the root valve to initially pressurize the sensing line but as the instrument drain valve leakage increased over time, it exceeded the leakage rate of the root valve and the sensing line pressure decreased to the annunciator setpoint. A root cause evaluation was performed using Nuclear Interface Procedure, NIP-ECA-02, "Root Cause Evaluations." The root cause was determined to be a lack of configuration management in that the instrument root valve had not been included in station prints or valve lineup procedures.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73 (a)(2)(i)(B).

10CFR50.73 (a)(2)(i)(B), "Any operation or condition prohibited by the plant's Technical Specifications." NMP1 Technical Specification table 3.6.2a requires 4 operable instrument channels per trip system for turbine stop valve closure and 2 for generator load rejection scram to be operable at greater than 45 percent reactor power.

During power operation above 45 percent reactor power, the Turbine Stop Valve (TSV) closure ($\leq$ 10 percent valve closure) and generator load rejection (loss of oil pressure to the acceleration relay) scrams are in operation. The scrams are provided to anticipate the rapid increase in reactor pressure and neutron flux as a result of a turbine trip and subsequent failure of the turbine bypass valves. These scrams ensure that the safety limit associated with Minimum Critical Power Ratio (MCPR) will not be exceeded. This is addressed in the Final Safety Analysis Report (FSAR), Section XV.

The turbine first stage bowl pressure corresponding to 45 percent reactor power is used to actuate annunciator F3-4-6, "Turbine First Stage Bowl Pressure Low" Alarm. At pressures less than annunciator setpoint, F3-4-6 is lit indicating TSV closure and generator load rejection scrams are bypassed. Below 45 percent reactor power, a turbine trip with failure to bypass will not result in exceeding MCPR limits.

This circuitry is a "one out of two taken twice" logic system. In order to satisfy the logic requirements to scram, one of two pressure switches in each trip system must be tripped. The failure of a single limit switch or single relay will not prevent the actuation of the RPS trip. The logic arrangement is such that pressure switches A and C actuate RPS trip system

III. ANALYSIS OF EVENT (cont.)

channel 11, and pressure switches B and D actuate RPS trip system channel 12. This event bypassed the scram capability of the A and B switches that provided inputs to both channel 11 and 12, thus resulting in less than the required number of operable instrument channels per operable trip system.

The pressure switches in the other sensing line were still functional to provide inputs to RPS channels 11 and 12. If a turbine trip occurred in this condition, a full scram would have been generated from either TSV position or loss of oil to the acceleration relay.

For the January 22nd event, Niagara Mohawk evaluated the limiting FSAR Chapter 15 transients at the plant conditions at the time of the event, assuming that the turbine stop valve closure and generator load rejection scrams would not have functioned. This evaluation determined that the MCPR safety limit would not have been violated had a transient occurred.

This event did not adversely compromise the safety of the plant, plant personnel or the general public.

IV. CORRECTIVE ACTIONS

The immediate corrective action was to respond to the alarming condition in accordance with station operating procedures. An additional corrective action was to increase reactor power to clear the alarming condition as soon as station operating conditions allowed.

The decision to increase power to clear the first stage bowl pressure low annunciator is appropriate for the following reasons:

- A small increase in reactor power was required to clear the alarm, thus reinstating the generator load rejection scram and turbine stop valve closure scram. Conditions that support this decision were the concurrent reverse flow operation (priority evolution and did not want to divert crew attention) and process computer being out of service. Without the process computer, plant Operations personnel could not verify alarms as required by annunciator response procedure.

IV. CORRECTIVE ACTIONS (cont.)

- The alternative would have been to reduce reactor power to below 45 percent. This would have taken significantly longer, leaving the plant without the required scram protection for a longer period of time. In addition, it would have been necessary to stop power reduction via decreasing recirculation flow just outside the restricted zone and to continue power reduction by inserting rods without the process computer. This would have hampered our ability to monitor core conditions.

Short Term Corrective Actions:

1. Implemented Temporary Modification (TM 5391), to disconnect the AC power leads for the pressure switch bypass relay which resulted in the scram functions being operational at all power levels.

2. Initiated DER# 1-92-Q-0125 to evaluate the condition, perform root cause evaluation and determine reportability.

3. Initiated WR# 201202 to investigate the instrumentation system.

4. Identified and opened the mispositioned root valve. This valve will be added to the applicable valve line-up, and will be added to station prints.

5. Performed Post Maintenance Testing for verification and returned system to operable status.

6. Review annunciators indicating bypassing of RPS functions for proper operator response and revise as required.

7. Verify the position of all root valves that affect Technical Specification instrumentation.

8. Senior Reactor Operators will review Limiting Safety System Settings (LSSS), including enabling and disabling Reactor Protection System circuitry.

Long Term Corrective Action:

1. Revise all appropriate documentation to reflect the results of the system walkdowns that are completed as part of the root valve verification program.

2. Repair/replace instrument drain valve (WR# 201028).

IV. CORRECTIVE ACTIONS (cont.

3. Plant personnel will be trained on the plant conditions requiring issuance of a Deviation Event Report and/or a Work Request.

V. ADDITIONAL INFORMATION

A. Failed components: none.

B. Previous similar events: none.

C. Identification of components referred to in this LER:

COMPONENT	IEEE 803 FUNCTION	IEEE 805 SYSTEM ID
Reactor Protection System	N/A	JC
Instrument Root Valve	RTV	JC
Instrument Drain Valve	DRN	JC
Intake Structure	N/A	NN
Annunciator F3-4-6 "Turbine First Stage Bowl Pressure"	ANN	SB
Turbine Stop Valve Scram Function	N/A	JC
Generator Load Rejection Scram Function	N/A	JC
Pressure Switches 02-13A, B, C, and D	63	JC
Circulating Water Intake Structure	N/A	NN