Niagara Mohawk

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NMP2L 1971

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

RE: Docket No. 50-410
LER 00-02, Supplement 1

Gentlemen:

In accordance with 10 CFR 50.73(a)(2)(iv), 10 CFR 50.73 (a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v), we are submitting Licensee Event Report 00-02, Supplement 1 "Manual Reactor Trip Due to an External Steam Leak on the Reactor Feedwater Pump and Automatic Trip of the RCIC System." This report provides the cause of this event, associated corrective actions, probabilistic risk analyis results, and information on NRC Form 366, Blocks 9, 10, and 12.

Very truly yours,

John T. Conway
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Vice President - Nuclear Generation

JTC/CES/tmk
Attachment

cc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Records Management

IE22

RGN-001

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION	APPROVED OMB NO. 3150-0104 EXPIRES:
LICENSEE EVENT REPORT (LER)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Nine Mile Point Unit 2	05000410	01 OF 06

TITLE (4)
Manual Reactor Trip Due to an External Steam Leak on the Reactor Feedwater Pump and Automatic Trip of the RCIC System

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)
03	03	00	00	02	01	06	15	00	N/A	
									N/A	

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)		
POWER LEVEL (10) 28%	☐ 20.2201(b)	☐ 20.2203(a)(2)(v)	☒ 50.73(a)(2)(i)	☐ 50.73(a)(2)(viii)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(i)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(iii)	☐ 73.71
	☐ 20.2203(a)(2)(ii)	☐ 20.2203(a)(4)	☒ 50.73(a)(2)(iv)	☐ OTHER *(Specify in Abstract below and in Text, NRC Form 366A)*
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(1)	☒ 50.73(a)(2)(v)	
	☐ 20.2203(a)(2)(iv)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Ray Dean - Manager Technical Support - Unit 2	(315) 349-4240

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES *(If yes, complete EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single space typewritten lines)* (16)

On March 3, 2000, at 1417 hours, while performing a normal plant shutdown, Nine Mile Point Unit 2 (NMP2) was manually tripped from 28 percent power. The plant was manually tripped due to a potential loss of the operating reactor feedwater pump 2FWS-P1B. The operators manually initiated the reactor core isolation cooling (RCIC) system in anticipation of losing the feedwater pump. The RCIC system automatically tripped on low suction pressure due to a partial void present in the discharge piping and the resultant water hammer.

Operators were concerned about losing the feedwater pump due to outboard seal injection low flow and high pressure indications and a steam leak. The cause of the incorrect flow and pressure indications was due to inservice wear. The cause of the steam leak was due to inadequate torquing of the discharge cover.

The cause of the RCIC system failure was that the design analysis assumed that steam leakage past the inboard testable check valve would condense and maintain the discharge piping downstream of the injection valve full of water. Contrary to this design assumption, there was no minimum leakage requirement or method of ensuring this assumption was met. The RCIC system piping and snubbers were inspected, the injection valve leak was fixed, and a keep-fill modification for the RCIC discharge piping is being implemented.

I. DESCRIPTION OF EVENT

On March 3, 2000, at 1417 hours, while performing a normal plant shutdown, NMP2 was manually tripped from 28 percent power. The operators tripped the reactor due to an external steam leak combined with seal injection flow and pressure indications on the operating reactor feedwater pump 2FWS-P1B.

The operators were performing a normal plant shutdown for a refueling outage, when an auxiliary operator reported that the operating feedwater pump had an external steam leak. Flow indicator 2FWS-FI12C for the outboard seal injection flow was indicating low, and Pressure Indicator 2FWS-PI3B for outboard seal injection pressure was out of range high. Direction was given to the control room operator to trip the reactor due to concerns with the operating reactor feedwater pump. The decision to trip the reactor was based on an external steam leak combined with the outboard seal injection low flow and high pressure indications.

An investigation team was formed to determine why the outboard seal injection indications were not reading normal and the reason for the steam leak. Internal damage was observed on the flow indicator and pressure indicator. The team determined that the damage to the flow indicator was induced wear due to extended operations at low flow conditions during plant startups and shutdowns and the damage to the pressure indicator was inservice wear that occurred during pump manipulations. Other similar flow and pressure indicators were inspected and no internal damage was identified. The lack of similar flow and pressure indication damage on the other two feedwater pumps was attributed to the higher amount of service time on Pump 2FWS-P1B, and associated instrumentation, accumulated during plant startups and shutdowns.

The team also identified that the steam leak was from the pump outboard discharge cover gasket. An inspection of the discharge cover gasket and corresponding mating surfaces revealed no visual damages. The team noted a discrepancy with the pump discharge cover torquing values. Two different maintenance procedures that governed work on the pump outboard discharge cover gasket contained lower torque values than recommended by the vendor. The torque values for the similar components on the other two feedwater pumps (2FWS-P1A and 2FWS-P1C) were either verified correct or torqued to the correct value.

Even with the noted problems discussed above, Feedwater Pump 2FWS-P1B remained in service and was used to complete the shutdown of the plant. The feedwater pump was secured over two hours later.

Following the reactor trip, the main generator tripped on reverse power as designed. No electrical system abnormalities occurred. The maximum reactor pressure recorded during the transient was 974 psig.

During the reactor trip, reactor water level reached a normal minimum of 151 inches (165.4 inches above the top of active fuel) and a maximum of 200 inches. Primary Containment Isolation Groups 4 (residual heat removal radwaste discharge and sampling valves) and 5 (residual heat removal shutdown cooling valves and other system valves) isolated due to reactor water level falling below the isolation setpoint of 159.3 inches (Level III). The Primary Containment Isolation Groups 4 and 5 valves were in their normal, closed position; therefore, the valves did not change position.

I. DESCRIPTION OF EVENT (Cont'd)

The operators initiated the RCIC system to maintain reactor water level in anticipation of a loss of feedwater flow. The RCIC pump started, but tripped 25 seconds after initiation. The operators declared the RCIC system inoperable. The operators closely monitored and kept the feedwater pump in service to maintain reactor water level due to the RCIC system trip.

In response to the RCIC system trip, a multi-discipline team was formed to investigate and determine the cause of the trip. The team identified that piping downstream of the RCIC system injection valve (2ICS*MOV126A) was voided due to a body-to-bonnet leak on the injection valve. Filling the void resulted in a water hammer event, which in turn resulted in a pressure wave. The pressure wave traveled back through the system into the suction piping causing the low suction pressure relay to actuate. The low suction pressure relay tripped the RCIC system, the operators reset the RCIC logic and placed the system in standby.

General Electric's design of the RCIC system assumes the discharge piping downstream of the injection valve will not be voided due to steam leakage past the inboard testable check valve which condenses to maintain the piping full of water. The inboard check valve does not have a minimum leakage requirement, but has a maximum leakage requirement. 10CFR50, Appendix J criteria limit the leakage across the inboard check valve to 0.78 gpm. General Electric's RCIC system design does not specify a minimum leakage requirement.

Therefore, if the inboard check valve is leak tight, condensed reactor steam is not available and then any leakage from the discharge piping volume can lead to voiding the pipe resulting in a water hammer and the RCIC system tripping.

NMPC reviewed past RCIC system problems, and concluded that voiding in the discharge piping downstream of the injection valve may have caused water hammer events in the past.

II. CAUSE OF EVENT

Operators were concerned about losing the feedwater pump due to outboard seal injection low flow and high pressure indications and a steam leak; therefore, the operators tripped the reactor. The cause of the incorrect flow and pressure indications was due to inservice wear, and the cause of the steam leak was due to inadequate torquing of the discharge cover.

The cause of the RCIC system failure was inadequate design analysis, in that General Electric's design assumed that leakage of condensed steam past the inboard check valve would maintain the piping downstream of the injection valve filled. Since the inboard check valve was leak tight, system leakage between the inboard check valve and the injection valve caused a void, resulting in a water hammer and a system trip on low suction pressure. Contributing to the RCIC system failure was a leak on the bonnet pressure seal of the injection valve, which resulted in the downstream piping being voided.

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

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73(a)(2)(iv), 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(v). 10CFR50.73(a)(2)(iv) requires a report when any event or condition resulted in a manual or automatic actuation of any engineered safety features, including the reactor protection system. The operators manually tripped the reactor. 10CFR50.73(a)(2)(i)(B) requires a report for any operation or condition prohibited by the plant's Technical Specifications. Technical Specification Surveillance Requirement 4.7.5.d requires an inspection of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients. NMPC concluded that the discharge piping downstream of the injection valve may have been voided and a water hammer event may have occurred in the past without the reactor core isolation cooling system snubbers being inspected as required. 10CFR50.73(a)(2)(v) requires a report when any event alone could have prevented the fulfillment of the safety function of a system to remove residual heat. The RCIC system tripped due to the discharge piping downstream of the injection valve being void of water which caused a water hammer and resulted in the RCIC system tripping on low suction pressure.

All control rods fully inserted in response to the reactor trip signal. The RCIC system failed to achieve rated flow. The feedwater system maintained flow and level. The high pressure core spray system was operable at the time of the event and is designed to inititate on a Level II signal (108.8 inches). The automatic depressurization system and the low pressure emergency core cooling systems were operable throughout this event.

The RCIC logic was reset and the system was returned to standby condition. The system was then available and would have performed its intended safety function if required, because there was a sufficient amount of water added to the system during the water hammer event to prevent the system from tripping on low suction.

The conditional core damage probability for this event has been analyzed using the Nine Mile Point Unit 2 probabilistic risk analysis model. The analysis assumes that the running feedwater pump will continue to operate, but assigned a higher than normal failure rate. No credit is given for the recovery of Feedwater Pump 2FWS-P1A. The analysis does recognize the potential for recovery of the reactor core isolation cooling system after 19 minutes from when the scram occurred. Based on the analysis, the conditional core damage probability is 6.4E-7.

Based on the information provided above, there were no adverse safety consequences as a result of this event. The reactor trip or the failure of the reactor core isolation cooling system posed no threat to the health and safety of the general public or plant personnel.

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

IV. CORRECTIVE ACTIONS

1. Niagara Mohawk Power Corporation (NMPC) reassembled Feedwater Pumps 2FWS-P1A and 2FWS-P1B using the vendor recommended gasket and torquing requirements.

2. NMPC reviewed and revised, where appropriate, the frequency of current preventive maintenance tasks performed on the feedwater pump instrumentation, and to include an internal inspection of the instruments regardless of calibration results.

3. NMPC revised Procedures N2-MPM-FWS-R133, "Reactor Feedwater Pumps (2FWS-P1A, 2FWS-P1B, 2FWS-P1C)" and N2-MMP-FWS-104, "Overhaul of Feedwater Pumps," to include the correct discharge cover torquing requirements.

4. NMPC completed a system walk down inspection. The walk down identified a broken suction pressure gage, which has been repaired.

5. NMPC repaired the body-to-bonnet pressure seal leak on the RCIC injection valve.

6. NMPC inspected the reactor core isolation cooling system snubbers, and did not identify any deficiencies.

7. NMPC implemented a keep-fill modification for the discharge piping of the reactor core isolation cooling system to preclude voiding.

V. ADDITIONAL INFORMATION

A. Failed components: none

B. Previous similar events:

- Licensee Event Reports 99-05, "Reactor Trip due to a Main Generator Protection Volts/Hertz Relay Failure," and 99-10, "Unit 2 Reactor Trip due to a Feedwater Master Controller Failure," documented failures of the reactor core isolation cooling system. Licensee Event Report 99-05 listed the cause to be an inadvertent trip of the trip throttle valve. After this condition (Licensee Event Report 00-02), NMPC reevaluated Licensee Event Report 99-05 and concluded that most likely the discharge piping downstream of the injection valve was voided and the water hammer may have contributed to the inadvertent trip of the throttle valve. As documented in Licensee Event Report 99-05, the plant transient analysis recorder failed to record data, and NMPC concluded the cause an inadvertent trip of the trip throttle valve based on the available data after the condition occurred. The cause identified in Licensee Event Report 99-10 was different than the cause for this condition. Therefore, the corrective actions described in these licensee event reports would not have prevented this condition from occurring.

NRC FORM 366A	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED OMB NO. 3150-0104 EXPIRES:
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 2	05000410	00	- 02	- 01	06 OF 06

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

V. ADDITIONAL INFORMATION (Cont'd)

C. Identification of components referred to in this license event report:

Components	IEEE 803A Function	IEEE 805 System ID
Reactor Feedwater Pump	P	SJ
Main Generator	GEN	EL
Reactor Core Isolation Cooling System	N/A	BN
Residual Heat Removal Isolation Valve	ISV	BO
Residual Heat Removal Radwaste Discharge Valve	V	BO
Residual Heat Removal Isolation Sample Valve	V	BO
Reactor Vessel	RPV	AD
Injection Valve	INV	BN
Pressure Gage	PI	BN
Piping	PSP	BN
Check Valve	V	BN
Reactor Core Isolation Cooling Pump	P	BN
Low Suction Pressure Relay	RLY	BN
Snubbers	SNB	BN
Bonnet Pressure Seal	SEAL	BN
Control Rods	ROD	AA
Flow Indicator	FI	SJ
Pressure Indicator	PI	SJ
Gasket	N/A	SJ
Seal	SEAL	SJ