Niagara Mohawk

April 3, 2000
NMP2L 1947

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

RE: Docket No. 50-410
LER 00-02

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, "Manual Reactor Trip Due to an External Steam Leak on the Reactor Feedwater Pump and Automatic Trip of the RCIC System."

Very truly yours,

Michael F. Peckham
Plant Manager - NMP2

MFP/SHC/cr
Attachment

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

IE22

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	1 OF 6

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	00	04	03	00	N/A	
									N/A	

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)			
POWER LEVEL (10)	☐ 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
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(1)	☒ 50.73(a)(2)(v)	*(Specify in Abstract below and in Text, NRC Form 366A)*
	☐ 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	(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		06	15	00

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.

Niagara Mohawk Power Corporation (NMPC) is continuing to investigate the cause of the external steam leakage of the 2FWS-P1B feedwater pump and will submit additional information and corrective actions in a supplement to this report.

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, Nine Mile Point Unit 2 was manually tripped from 28 percent power. The operators tripped the reactor due to an external steam leak 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. The station shift supervisor directed the control room operator to trip the reactor.

NMPC is continuing to investigate the cause of the external steam leakage of the 2FWS-P1B feedwater pump and will submit additional information and corrective actions in a supplement to this report.

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.

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

DESCRIPTION OF EVENT (Cont'd)

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.

CAUSE OF EVENT

Niagara Mohawk Power Corporation is continuing to investigate the cause of the external steam leakage of the 2FWS-P1B feedwater pump and will submit additional information and corrective actions in a supplement to this report.

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.

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.

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Nine Mile Point Unit 2	0 5 0 0 0 4 1 0	0 0	-	0 2	-	0 0	04 OF 06

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

ANALYSIS OF EVENT (Cont'd)

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.

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.

CORRECTIVE ACTIONS

1. NMPC will submit the corrective actions for the reactor feedwater pump external steam leakage in a supplement to this report by June 15, 2000.

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

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

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

5. NMPC will be implementing a keep-fill modification for the discharge piping of the reactor core isolation cooling system to preclude voiding prior to April 15, 2000.

ADDITIONAL INFORMATION

Failed components: none

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.

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

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