CATEGORY

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FACIL:50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410
AUTH.NAME AUTHOR AFFILIATION
BOSNIC,D.P. Niagara Mohawk Power Corp.
PALEOLOGOS,N. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 99-005-00:on 990424,RT was noted.Caused by main generator protection volts/hertz relay failure.Stablized plant,determined & corrected cause of equipment failure, revised procedures & trained personnel.With 990524 ltr.

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A04

Niagara Mohawk

May 24, 1999
NMP2L 1869

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

RE: Docket No. 50-410
LER 99-05

Gentlemen:

In accordance with 10CFR50.73(a)(2)(i)(B), 10CFR50.73(a)(2)(iv), 10CFR50.73(a)(2)(vii), and Technical Specification 3.5.1.f, we are submitting LER 99-05, "Reactor Trip Due to a Main Generator Protection Volts/Hertz Relay Failure."

Very truly yours,

Nick Paleologos

Nick Paleologos
Plant Manager - NMP2

NCP/CES/kap
Attachment

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

IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES:

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 10

TITLE (4)
Reactor Trip Due to a Main Generator Protection Volts/Hertz Relay Failure

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)
04	24	99	99	05	00	05	24	99	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): 100%

☐ 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)	Special Report

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Don P. Bosnic - Operations Manager	315-349-7952

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
D	BN	SCV	G153	YES	X	SA	V	B045	YES
B	EF	FU	G187	YES	X	EL	RLY	G082	YES

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 April 24, 1999, Nine Mile Point Unit 2 experienced an automatic reactor scram from 100 percent power. The cause of the reactor scram was a fast closure of the main turbine stop and control valves. Operators responding to the event were challenged by several equipment concerns including: failure of the reactor core isolation cooling system to achieve rated flow, the trip of two electrical protection assemblies, an air leak on an auxiliary boiler valve, and the need to restart numerous pieces of plant equipment due to the loss of electrical power during the residual (slow) transfer of the non-vital 13.8 kV buses.

The cause of the fast closure of the turbine stop and control valves was determined to be a failure of a main generator protection volts/hertz relay.

Corrective actions included: stabilizing the plant, determining and correcting the cause of the equipment failures, revising procedures, training personnel and establishing longer term corrective actions.

Technical Specification 3.5.1.f requires a special report when an emergency core cooling system injects water into the reactor coolant system. During this event the high pressure core spray system injected water into the reactor vessel.

I. DESCRIPTION OF EVENT

On April 24, 1999, Nine Mile Point Unit 2 automatically tripped from 100 percent power. The cause of the reactor trip was a fast closure of the turbine stop and control valves. There were no maintenance or testing activities in progress at the time of the event.

Niagara Mohawk Power Corporation (NMPC) determined that the cause of the fast closure of the turbine stop and control valves was a volts/hertz relay failure, which actuated a generator lockout relay. The plant information recorders indicated that the first alarm point was the generator lockout relay, followed by the turbine trip signal. Actuation of either the volts/hertz relay or the neutral over-current relay could have caused the generator lockout relay to actuate. The neutral over-current relay tested satisfactory. However, the volts/hertz relay did not test satisfactorily, and this confirmed that it was the event initiator. The volts/hertz relay was replaced with a new relay less than one year prior to its failure.

When the generator lockout relay actuated, it initiated an automatic residual (slow) transfer of the 13.8 kV non-safety related busses to off-site power sources. The electrical protection scheme tripped large electrical loads from the 13.8 kV busses as designed to prevent damaging plant equipment during the reenergization of switchgear. These large loads included the reactor feedwater pumps, reactor recirculation pumps, and condensate booster pumps.

During the residual transfer, electrical power was momentarily lost to some of the normal lighting. The lighting affected was in portions of the turbine, reactor, normal switchgear, screenwell, and control buildings. Also, some of the security perimeter lighting was momentarily interrupted. The transient had no effect on essential or emergency lighting. The lighting system responded to the transient as designed and did not effect the operator's response to the transient.

Coincident with the residual transfer, was a loss of output voltage from an uninterruptible power supply which provided a portion of the logic power to the reactor protection system. The most probable cause of the uninterruptable power supply malfunction was a design deficiency in the maintenance bypass transfer switch which caused the inverter input fuse to blow. The maintenance bypass transfer switch, not part of the uninterruptible power supply, was installed to feed loads from another alternating current source while performing maintenance on the uninterruptible power supply. The uninterruptable power supply blown fuse was most probably caused by electrical noise generated at the maintenance bypass transfer switch neutral terminal which is connected as a reference point to the inverter control circuits. High circulating current between the inverter and maintenance bypass power output was most probably caused when the plant 13.8 kV system voltage decayed and the inverter operated off the direct current input. The circulating current burned the circuit board trace, which is in close proximity of the neutral terminal. The trace arced, thereby inducing electro-magnetic interference/radio frequency interference noise into the neutral terminal and impacting the operation of the uninterruptible power supply silicon controlled rectifiers firing mechanism which caused two silicon controlled rectifiers to fire simultaneously. This blew the inverter input direct current fuse, resulting in the loss of alternating current output voltage, and initiated the timers on the electrical protection assemblies.

I. DESCRIPTION OF EVENT (Cont'd)

The timers in the electrical protection assemblies timed out, causing the two electrical protection assemblies to trip before maintenance supply power was restored to the uninterruptible power supply.

During the reactor trip, reactor water level reached a minimum of 105 inches (119.4 inches above the top of active fuel) and a maximum of 202 inches. Primary containment isolation occurred either due to reactor water level falling below isolation setpoints of 159.3 (Level III) and 108.8 (Level II) inches or the loss of two electrical protection assemblies. NMPC could not determine which signal came in first. The signals consisted of the following groups:

Group 2	Isolation signals to the reactor water outboard sample line isolation valves.
Group 3	Isolation signal to the nitrogen purge isolation valve to the transversing in-core probe.
Group 4	Isolation signal to the residual heat removal system sample line valves.
Group 5	Isolation signal to the residual heat removal system shutdown cooling suction valves.
Group 6	Isolation signal to the reactor water cleanup outboard isolation valve.
Group 7	Isolation signal to the reactor water cleanup inboard isolation valve.
Group 8	Isolation signal to the reactor building closed loop cooling water, drywell fire protection, automatic depressurization system air lines, instrument air, containment leakage monitoring, and reactor recirculation hydraulic power unit lines.
Group 9	Isolation signal to primary containment purge system isolation valves.

The plant response to the various containment isolation signals was as expected for the plant design.

Several high drywell temperatures alarms were received because of the isolation of drywell cooling. The maximum average drywell temperature reached was 118 degrees Fahrenheit. Operators took appropriate actions to restore drywell cooling, resulting in the temperature returning to normal. NMPC evaluated the drywell transient and peak temperature data and determined that the increased temperatures were bounded by the equipment qualification.

The high pressure core spray system automatically initiated and performed as designed to initially control reactor water level. After operators started a condensate booster pump and feedwater pump, operators controlled reactor water level with the condensate and feedwater system.

I. DESCRIPTION OF EVENT (Cont'd)

The reactor core isolation cooling system initiated, as a result of reactor water level descending below initiation setpoint of 108.8 (Level II), but failed to achieve rated speed. Control room operators observed that the automatic initiation sequence started, and the steam inlet valve and the injection valve opened, but the trip throttle valve indicated intermediate position. The operators confirmed that the high pressure core spray system was controlling reactor water level, then manually tripped the turbine as an equipment protection measure in accordance with operating procedures. After the manual trip, the trip throttle valve position continued to indicate intermediate and the trip alarm was not received for reactor core isolation cooling. The operators then manually closed the trip throttle valve and the steam inlet valve and declared the reactor core isolation cooling system inoperable.

Troubleshooting efforts revealed two problems. The first problem was that the trip throttle valve was in the closed position, but the closed limit switch was not actuated. This limit switch prevented the trip alarm from annunciating, and resulted in intermediate position indication. The second problem was that the trip throttle valve overspeed trip mechanism was incorrectly adjusted for adequate engagement of the trip hook and latch up lever during Refueling Outage 6. Inadequate engagement coupled with vibration from the steam admission on the initiation signal caused the trip throttle valve to unlatch and close. During troubleshooting with the trip throttle valve open, a mechanic touched the clevis and pin at the connecting rod to the mechanical trip leakage slightly, and this caused the trip throttle valve to unlatch and close. NMPC reviewed the Refueling Outage 6 work order for the overspeed trip mechanism and determined that the work order and Procedures N2-MMP-ICS-244, "Maintenance of Reactor Core Isolation Cooling Turbine Trip and Throttle Valve", and N2-MPM-ICS-V452, "Reactor Core Isolation Cooling Turbine and Accessories", did not provide proper instructions for the adjustment of the overspeed trip rod, the trip hook and latch up lever engagement, and the spring tension. A review of the vendor manual identified that the manual did include setup information for the overspeed trip mechanism. NMPC determined that the reactor core isolation cooling system was potentially inoperable since the last quarterly surveillance performed on February 20, 1999, when steam flowed through the trip throttle valve and the valve demonstrated proper operation. This is based on the fact that when the reactor core isolation cooling system initiated, the reactor pressure was approximately normal operating pressure. Therefore, NMPC has concluded that the reactor core isolation cooling system may not have responded as designed at any time subsequent to the last successful surveillance test.

The maximum reactor pressure recorded during the transient was 1088 psig. This is 15 psig below the lowest actuation setpoint for the main steam safety relief valves. The safety relief valves did not open. The main steam isolation valves remained open throughout the event, and all five turbine bypass valves opened to control reactor pressure by directing steam to the main condenser.

The plant transient analysis recorder failed to trip and record reactor scram data because the trip setpoints were improperly set following troubleshooting earlier in the shift. This complicated data gathering for

I. DESCRIPTION OF EVENT (Cont'd)

the investigation of the transient. Deviation/Event Report 2-1999-1260 was initiated to determine the cause and develop corrective actions.

Later in the event, operators started a condensate booster pump and determined that the pump could not be used to maintain reactor water level with reactor pressure at 600 psig. This deviated from plant response modeled in the simulator which suggested that the condensate booster pump could begin injecting water at approximately 650 psig. NMPC will review the training scenarios and the transient and determine if the simulator model needs to be revised (Deviation/Event Report 2-1999-1287). Upon recognition that the condensate booster pump was ineffective in raising reactor water level, the operators promptly started a feedwater pump in accordance with operating procedures.

There were a number of problems that required operators to take additional action to recover from the transient. The operators had to start various pieces of equipment due to load shedding as a result of the loss of electrical power during the residual transfer (for example, feedwater pump, condensate pump, instrument air compressors, turbine lift pumps and ventilation systems). While these equipment issues added some complexity to the operators' actions, operators effectively managed the event and placed the plant in cold shutdown. Additionally:

- An auxiliary boiler valve had an air leak, which resulted in the operators stopping the attempt to start one auxiliary boiler and starting the other auxiliary boiler. Work Order 99-08040-00 was written to repair the valve.

- The Main Steam Line B and D radiation monitors indicated greater than 4,000 mREM on the safety parameter display system, with the actual reading on the radiation monitor panel reading normal (1 mREM). NMPC determined that the cause was a loss of power due to the tripping of the electrical protection assembly. Deviation/Event Report 2-1999-1270 was written to incorporate this problem into the corrective action program.

II. CAUSE OF EVENT

The cause of the reactor trip was determined to be a failure of a volts/hertz relay. The relay has been sent out for further failure analysis.

The most probable cause of the Uninterruptible Power Supply 2VBB-UPS3B trip was a design deficiency in the maintenance bypass transfer switch logic controls. As a result of the design deficiency, noise was generated on the maintenance bypass transfer switch neutral terminal which subsequently caused the uninterruptible power supply inverter fuse to blow.

II. CAUSE OF EVENT (Cont'd)

The cause of the reactor core isolation cooling system failure was determined to be an inadvertent trip of the trip throttle valve. The overspeed trip mechanism was incorrectly aligned. The work order and reactor core isolation cooling system procedures performed during Refueling Outage 6 did not sufficiently address the setup of the overspeed trip mechanism consistent with the vendor manual.

III. ANALYSIS OF EVENT

This event is considered reportable under 10CFR50.73(a)(2)(i)(B), 10CFR50.73(a)(2)(iv), 10CFR50.73(a)(2)(vii), and Technical Specification 3.5.1.f. 10CFR50.73(a)(2)(i)(B) requires a report for any operation or condition prohibited by the Technical Specification. Technical Specification 3.7.4 was potentially not met for the reactor core isolation cooling system due to inadequate maintenance performed on the trip throttle valve overspeed trip mechanism. 10CFR50.73(a)(2)(iv) requires a report for any event or condition that resulted in manual or automatic actuation of any engineered safety features, including the reactor protection system. 10CFR50.73(a)(2)(vii) requires a report when any event caused at least one independent train to become inoperable. Technical Specifications 3.5.1.f requires a special report when an emergency core cooling system injects water into the reactor coolant system. The high pressure core spray system injected water into the reactor vessel. The total accumulated initiation cycles is 9, and the current usage factor value remains less than 0.70.

The reactor trip is the design response for fast closure of the main turbine stop and control valves. All control rods fully inserted after the reactor trip signal. The reactor core isolation cooling system failed to achieve rated flow, and therefore was not used for level control. The high pressure core spray system initiated and maintained reactor water level as designed. The automatic depressurization system and low pressure emergency core cooling systems were operable throughout this event. After the residual transfer was completed, the operators restarted and used a condensate booster pump and feedwater pump to maintain reactor water level.

The conditional core damage probability for this event was calculated to be 8.6E-6. The conditional core damage probability was determined based on the loss of feedwater as the initiating event with possible recovery. The reactor core isolation cooling system was conservatively assumed to be lost for the event.

The plant response was in accordance with the Updated Safety Analysis Report transient analysis for a generator load reject with bypass valve operation with the exception of the reactor core isolation cooling system and uninterruptible power supply failures.

Based on the above analysis, there were no adverse safety consequences as a result of this event. The reactor trip posed no threat to the health and safety of the general public or plant personnel.

IV. CORRECTIVE ACTIONS

1. Operators performed scram recovery actions, and placed the plant in a stable condition.

2. The volts/hertz relay was replaced, and a failure analysis will be performed on the failed relay.

3. The uninterruptible power supply blown fuse was replaced, the uninterruptible power supply was inspected for internal damage, and the motor operated feature of the maintenance bypass transfer switches which contained the design deficiency was disabled.

4. Deviation/Event Report 2-1999-1707 was written to determine the cause of the design deficiency of the maintenance bypass transfer switch and to determine appropriate long-term preventive actions.

5. The overspeed trip mechanism for the reactor core isolation cooling trip throttle valve was adjusted using Work Order 99-08055-05 and was tested at 150 psig reactor pressure and at rated reactor pressure.

6. The reactor core isolation cooling trip throttle valve closed limit switch was replaced and tested satisfactorily.

7. Maintenance reviewed all preventative and corrective maintenance on the reactor core isolation cooling turbine, governor valve, the overspeed trip mechanism, and lube oil system during and subsequent to Refueling Outage 6 to ensure that work was performed to the design criteria, that procedure requirements coincided with vendor manual requirements, and that the steps performed met the acceptance criteria. No other problems were identified during this review.

8. Operators were given training and procedural guidance to ensure the reactor core isolation cooling trip throttle valve is properly latched following activities that unlatch the trip throttle valve.

9. Applicable maintenance procedures for the reactor core isolation cooling system will be revised to provide specific instructions to align the overspeed trip mechanism by July 31, 1999.

10. Deviation/Event Report 2-1999-1723 was written to determine the cause of the inadequate work instructions for the reactor core isolation cooling overspeed trip mechanism and to initiate corrective actions.

V. ADDITIONAL INFORMATION

A. Failed components:

- The volts/hertz relay failed on April 24, 1999, which was the cause of the transient. The volts/hertz relay was considered operable based on the relay performance prior to its failure.

- The reactor core isolation cooling trip throttle valve was potentially inoperable since the last quarterly surveillance performed on February 20, 1999, when steam flowed through the trip throttle valve and the valve demonstrated proper operation.

- The uninterruptible power supply fuse failed at the time of the residual transfer of the 13.8 kV bus on April 24, 1999.

- The auxiliary boiler was inoperable when operators attempted to place the boiler in service because of a valve air leak. NMPC could not determine when the air leak developed.

B. Previous similar events:

Nine Mile Point Unit 2 has had a number of instances where engineered safety feature actuations occurred (Licensee Event Reports 98-13, 98-06, 98-05, 97-04, and 96-04). The root causes of the licensee event reports were different than the root cause for this event. Therefore, the corrective actions from these licensee event reports would not have prevented this engineered safety feature actuation from occurring.

There have been no recent similar failures of the reactor core isolation cooling system.

C. Identification of components referred to in this LER:

COMPONENT	IEEE 803A FUNCTION	IEEE 805 SYSTEM ID
Volts/Hertz Relay	RLY	EL
Generator Lockout Relay	RLY	EL
Neutral Over-Current Relay	RLY	EL
Reactor Core Isolation Cooling Turbine	TRB	BN

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Nine Mile Point Unit 2	05000410	99	- 05	- 00	09 OF 10

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

V. ADDITIONAL INFORMATION (Cont'd)

COMPONENT	IEEE 803A FUNCTION	IEEE 805 SYSTEM ID
Reactor Core Isolation Cooling Trip Throttle Valve	SCV	BN
Reactor Core Isolation Cooling Steam Inlet Valve	SHV	BN
Reactor Core Isolation Cooling Injection Valve	INV	BN
Primary Containment Isolation Valves	ISV	NH
Reactor Water Cleanup Isolation Valve	ISV	CE
Reactor Water Sample Line Isolation Valve	ISV	CE
Nitrogen Purge Isolation Valve	ISV	LK
Residual Heat Removal Shutdown Cooling Valve	ISV	BO
Residual Heat Removal Isolation Valve	ISV	BO
Fire Protection Isolation Valve	ISV	KP
Reactor Building Closed Loop Isolation Valve	ISV	CC
Instrument Air Isolation Valve	ISV	LD
Containment Leakage Monitoring Isolation Valve	ISV	IJ
Primary Containment Purge Isolation Valve	ISV	LK
Reactor Recirculation Hydraulic Power Unit Isolation Valve	ISV	AD
Reactor Recirculation Pump	P	AD
High Pressure Core Spray Nozzles	NZL	BG
Uninterruptible Power Supply	UJX	EF
Fuse	FU	EF
Electrical Switchgear	SWGR	EA
Main Turbine	TRB	TA
Turbine Stop and Control Valve	SHV	TA
Turbine Lift Pumps	P	TA
Turbine Bypass Valve	V	JI
Safety Relief Valves	RV	SB
Main Steam Isolation Valves	ISV	SB
Limit Switch	33	BN
Reactor Feedwater Pumps	P	SJ
Condensate Booster Pumps	P	SD

NRC FORM 366A   U.S. NUCLEAR REGULATORY COMMISSION

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

V. ADDITIONAL INFORMATION (Cont'd)

COMPONENT	IEEE 803A FUNCTION	IEEE 805 SYSTEM ID
Electrical Protection Assembly	N/A	EF
Silicon Controlled Rectifiers	SCR	EF
Plant Lighting	N/A	N/A
Instrument Air Compressors	CMP	LD
Radiation Monitor	MON	IL
Auxiliary Boiler Valve	V	SA
Plant Transient Analysis Recorder	XR	IQ