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 AUTH. NAME AUTHOR AFFILIATION
 CONWAY, J.T. Niagara Mohawk Power Corp.
 DAHLBERG, K.A. Niagara Mohawk Power Corp.
 RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 95-003-00: on 950214, momentary loss of Division I emergency 125 volt dc power resulted in trip of reactor recirculation pumps, causing manual reactor scram per plant operating procedures. Restored DC bus. W/950316 ltr.

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March 16, 1995
NMP2L 1534

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

RE: Docket No. 50-410
LER 95-03

Gentlemen:

In accordance with 10CFR50.73 (a)(2), we are submitting LER 95-03, "Manual Reactor Scram and ESF Actuations When Both Reactor Recirculation Pumps Tripped Because of a Loss of Emergency DC Power."

A telephone report of this event was made in accordance with 10CFR50.72 (b)(2)(ii) at 0058 hours on February 15, 1995.

Very truly yours,



Kim A. Dahlberg
Plant Manager - NMP2

KAD/RLM/lmc
Attachment

xc: Mr. Thomas T. Martin, Regional Administrator
Mr. Barry S. Norris, Senior Resident Inspector

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LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 500 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.

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TITLE (4) **Manual Reactor Scram and ESF Actuations When Both Reactor Recirculation Pumps Tripped Because of a Loss of Emergency DC Power**

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)		
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0 2	1 4	9 5	9 5	0 0 3	0 0 0	3 1 6	9 5		N/A			0 5 0 0 0		

OPERATING MODE (8) 1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)									
POWER LEVEL (10) 110 0	<input type="checkbox"/> 20.402(b)	<input type="checkbox"/> 20.405(c)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)	<input type="checkbox"/> 73.71(b)						
	<input type="checkbox"/> 20.405(a)(1)(i)	<input type="checkbox"/> 50.38(c)(1)	<input type="checkbox"/> 50.73(a)(2)(v)	<input type="checkbox"/> 73.71(c)						
	<input type="checkbox"/> 20.405(a)(1)(ii)	<input type="checkbox"/> 50.38(c)(2)	<input type="checkbox"/> 50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)						
	<input type="checkbox"/> 20.405(a)(1)(iii)	<input type="checkbox"/> 50.73(a)(2)(i)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)							
	<input type="checkbox"/> 20.405(a)(1)(iv)	<input type="checkbox"/> 50.73(a)(2)(ii)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)							
	<input type="checkbox"/> 20.405(a)(1)(v)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(x)							

LICENSEE CONTACT FOR THIS LER (12)

NAME John T. Conway, Manager Operations NMP2	TELEPHONE NUMBER
	AREA CODE: 3 1 5 NUMBER: 3 4 9 - 2 6 9 8

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)

YES (If yes, complete EXPECTED SUBMISSION DATE) | NO

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

On February 14, 1995 at 2112 hours, Nine Mile Point Unit 2 (NMP2) experienced a momentary loss of Division I Emergency 125VDC power. This resulted in a trip of both Reactor Recirculation pumps, causing control room operators to initiate a manual reactor scram per plant operating procedures. The momentary loss of Division I Emergency 125VDC power also caused isolation of Primary Containment Isolation System groups 8 and 9 and the normal Reactor Building (Secondary Containment) ventilation system. At the time of the event, the reactor mode switch was in the "RUN" position (Operational Condition 1) with the plant operating at approximately 100 percent of rated thermal power.

The cause of this event was personnel error caused by inadequate self-verification when the operator transferring battery chargers supplying the Emergency 125VDC bus opened the wrong breaker.

Immediate corrective actions included restoring the DC bus, completing reactor scram immediate actions, and stabilizing the plant in a hot shutdown condition. Additional corrective actions include: 1) counseling and disciplining the operator involved; 2) redefining and training operators on self-verification techniques; 3) modifying the operator training program; 4) reinforcing performance monitoring expectations with shift supervision; and 5) adding cubicle numbers to cubicle labels and revising procedural descriptions to agree with the new labels.



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.

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

I. DESCRIPTION OF EVENT

On February 14, 1995 at 2112 hours, Nine Mile Point Unit 2 (NMP2) experienced a momentary loss of Division I Emergency 125VDC power. This resulted in a trip of both Reactor Recirculation pumps, causing control room operators to initiate a manual reactor scram per plant operating procedures. The momentary loss of Division I Emergency 125VDC power also caused isolation of Primary Containment Isolation System (PCIS) groups 8 and 9 and the normal Reactor Building (Secondary Containment) ventilation system. At the time of the event, the reactor mode switch was in the "RUN" position (Operational Condition 1) with the plant operating at approximately 100 percent of rated thermal power.

A flush of the area unit cooler located above battery charger 2BYS*CHGR2A1 was planned for February 15, 1995. Because the flush involves temporary hose connections to the unit cooler and the potential for water leakage to impact battery charger operation, it was decided to place battery charger 2BYS*CHGR2A2 in service and remove 2BYS*CHGR2A1 from service. This evolution involves removing the operating battery charger from service, relocating its output breaker and starting the standby battery charger. The battery charger transfer was discussed at the pre-shift briefing, including the potential adverse consequences of a loss of Division 125VDC power. Prior to the evolution, a copy of the appropriate sections of Operating Procedure N2-OP-74A, "Emergency DC Distribution," was obtained by the operator performing the task and reviewed with the Chief Shift Operator (CSO). The operator performing the task then went to the switchgear room and walked-through the procedure locating all the equipment to be operated. During actual task performance, the operator properly deenergized operating battery charger 2BYS*CHGR2A1. At 2112 hours the operator incorrectly pressed the trip button for breaker 1B instead of breaker 2B, disconnecting battery 2BYS*BAT2A from the Emergency 125VDC bus 2BYS*SWG002A, causing a loss of Division I Emergency 125VDC power. The operator checked the bus indicators to verify his actions were correct and noticed zero volts DC indicating on bus 2BYS*SWG002A. The operator then realized his mistake and reclosed breaker 1B reconnecting 2BYS*BAT2A to the bus.

The loss of bus 2BYS*SWG002A caused a trip of one fast speed supply breaker for each of the Reactor Recirculation pumps. Since both pumps were running in fast speed, both pumps tripped off. The control room licensed operators observed annunciators and realized both Reactor Recirculation pumps had tripped. The Assistant Station Shift Supervisor (ASSS) directed the mode switch be placed in "Shutdown" and the scram was initiated at 2113 hours. The plant responded as expected and none of the safety-relief valves lifted. Level shrink caused by the rapid down power transient caused Reactor Pressure Vessel (RPV) level to drop below Level 3 (159.3 inches) to a low of 149 inches indicated (which is 163.4 inches above the top of active fuel). The level drop caused an automatic low water level reactor scram signal and an automatic isolation of PCIS groups 4 (Residual Heat Removal System sample valves) and 5 (Residual Heat Removal shutdown cooling and head spray isolation



**LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION**

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I. DESCRIPTION OF EVENT (Cont'd.)

valves). Also at Level 3, the control room operators entered Emergency Operating Procedure N2-EOP-RPV, "RPV Control."

Along with the loss of Reactor Recirculation pumps, the loss of Division I Emergency 125VDC power caused isolation of PCIS groups 8 (Primary Containment Auxiliary Systems isolation valves) and 9 (Primary Containment Purge System isolation valves) along with isolation of the normal Reactor Building (Secondary Containment) ventilation.

The immediate actions taken by the operators included verifying proper plant response to the scram signal, restarting the Reactor Recirculation pumps in slow speed, restoring RPV level using the Feedwater system to allow exiting N2-EOP-RPV, and stabilizing the plant in hot shutdown (Operational Condition 3). The operators also reset the PCIS isolation and reactor scram signals.

II. CAUSE OF EVENT

A root cause analysis for this event was performed in accordance with Nuclear Interfacing Procedure NIP-ECA-01, "Deviation Event Report." Because of the significance of the event, an additional barrier analysis was performed independent of the root cause analysis.

The root cause of this event was determined to be personnel error due to inadequate self-verification to ensure the correct component was manipulated. Specifically, the operator had verified the correct components while "walking through" the procedure. When actually performing the evolution, the operator did not properly verify his actions and he pressed the incorrect trip button, opening the wrong breaker.

Two contributing causes were also identified.

1. Lack of concurrent verification has been identified in previous events involving component mispositioning. However, actions to implement increased concurrent verification have not been formalized nor consistently implemented.
2. The human factor aspects of switchgear labeling and associated procedural text also contributed. Individual cubicles have installed nameplates but cubicle numbers are not typically included. As a result, a second means of identifying correct breakers is not present on most switchgear labels.



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II. CAUSE OF EVENT (Cont'd.)

The barrier analysis performed also identified four barriers that, if strengthened, may prevent similar future events.

- Equipment design did not prevent the adverse consequences of the error. This could be accomplished by coloring or configuring supply breakers differently from load breakers, supplying a separate breaker for each battery charger and allowing battery chargers to be paralleled while rotating them.
- The pre-job briefing did not prevent the error in that the operator was not fully sensitized to the risk and consequences associated with opening the wrong breaker. As a result, special precautions to prevent this error were not taken.
- Verbalization of actions. The self-verification work practice could benefit if each procedural step was verbalized while comparing the step to the component label.
- Supervisory skills for coaching/counseling subordinates work practices did not prevent the self-verification error. Shift supervision is not only responsible for safe operation of the plant and direction of the workforce but to provide coaching and counseling when required. Effective monitoring is necessary in order that appropriate, timely action is taken when expectations are not met. In many cases, shift supervision has not been afforded the supervisory training needed to develop the desired monitoring, coaching and counseling skills.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73 (a)(2)(iv), "any event or condition that resulted in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)."

The manual reactor scram is a conservative action to the loss of reactor recirculation flow which prevents thermal-hydraulic instabilities. This event was bounded by the Two Recirculation Pump Trip event analyzed in Section 15.3.1 of the NMP2 Updated Safety Analysis Report (USAR). The scenario in the USAR assumes that a high RPV level trip of the Feedwater System pumps and main turbine causes the reactor scram. In this event, the Feedwater Control System response avoided the high RPV water level trip reducing the severity of the transient. This is consistent with the initial startup test program results. The manual scram was initiated to prevent the development of thermal hydraulic instabilities in accordance with current industry practice.



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III. ANALYSIS OF EVENT (Cont'd.)

The analysis of the event discovered that the indicated natural circulation flow at the time of the scram was slightly less than the lowest flow characteristic shown on the power/flow map used to guide operation of the unit. The deviation between indicated core flow and the power/flow map was deemed acceptable for several reasons:

1. The difference (approximately 28 percent indicated versus 29 percent on the map) is within the measurement accuracy of the core flow instruments.
2. The "Natural Circulation Low Recirc Pump Speed both FCVs at Min. Position" core flow line on the power/flow map is a best estimate description for both natural circulation and low recirculation pump speed with the Flow Control Valves at their minimum position. The curve is based upon a limited set of data points collected during the initial Startup Test Program for the unit.
3. The core flow line on the power/flow map cannot take into account all the variables that affect natural circulation flow (e.g., power distribution, crudding of flow paths, and reactor vessel water level).

In conclusion, normal operations at natural circulation core flow are not planned for NMP2 but is within the bounds of the plant safety analysis. Also, due to the variables that affect natural circulation core flow, the curve should be interpreted only as a reasonable approximation of actual conditions.

The PCIS and Reactor Building normal ventilation system isolations were also conservative actions relative to the potential loss of coolant indicated by low RPV level.

This event occurred at approximately 100 percent rated thermal power and at no time was the ability of the operators to achieve and maintain safe plant conditions jeopardized. Therefore, there was no adverse impact on the health and safety of the general public or plant personnel. The event duration from the loss of Division I Emergency 125VDC power until the scram was reset was 32 minutes.

IV. CORRECTIVE ACTIONS

The immediate corrective actions included reenergizing the Division I Emergency 125VDC bus, implementation of reactor scram immediate actions, following N2-EOP-RPV actions for low RPV water level and stabilizing the plant in a hot shutdown condition.



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IV. CORRECTIVE ACTIONS (Cont'd.)

Significant corrective actions include:

1. The operator involved in this event was counseled on the proper method of self-verification and disciplinary action was taken.
2. An improvement plan, tailored to the operator involved was developed and implemented by the operator's shift supervision to reinforce good work practice habits during in-plant evolutions.
3. Operations Management will evaluate and clarify the proper self-verification technique for component manipulations.
4. The Operator Training Program will be modified to include detailed initial and continued training on self-verification and procedural place-keeping. The training will include evaluated Job Performance Measures (JPMs) covering Control Room and in-plant evolutions and will be required for both licensed and non-licensed operators.
5. Shift supervision will have performance monitoring expectations reinforced with monitoring provided by Operations Management.
6. Operations Management will determine and formalize expectations for times when concurrent verification is required.
7. All switchgear cubicles will have the cubicle number added to their labels and the Electrical Distribution Operating Procedures will be revised to ensure breaker nomenclature agrees with the cubicle label.

V. ADDITIONAL INFORMATION

- A. Failed components: none.
- B. Previous similar events: none.



**LICENSEE EVENT REPORT (LER)
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V. ADDITIONAL INFORMATION (Cont'd.)

C. Identification of components referred to in this LER:

COMPONENT	IEEE 803 EIS FUNCTION	IEEE 805 SYSTEM ID
Emergency 125VDC Power	N/A	EJ
Primary Containment Isolation System	N/A	BD
Reactor Building Ventilation System	N/A	VA
Reactor Recirculation System	N/A	AD
Reactor Protection System	N/A	JC
Residual Heat Removal System	N/A	BO
Primary Containment Purge System	N/A	VB
Feedwater System	N/A	SJ
Reactor Pressure Vessel	N/A	SB
Reactor Water Cleanup System	N/A	CE
Unit Cooler	CLR	VA
Breaker	BKR	EJ
Battery	BTRY	EJ
Pump	P	AD
Safety-Relief Valves	RV	SB
Valves	V	BO
Battery Charger	BYC	EJ
Reactor Mode Switch	HS	JC



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