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Docket No. 50-461

10CFR50.73

Document Control Desk
Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Clinton Power Station - Unit 1
Licensee Event Report No. 96-008-00

Dear Sir:

Enclosed is Licensee Event Report No. 96-008-00: Loose Terminal Connection Causes Reactor Recirculation Pumps to Trip from Fast to Slow, Operation in the Restricted Zone and Manual Scram. This report is being submitted in accordance with the requirements of 10CFR50.73.

Sincerely yours,

Wilfred Connell

Wilfred Connell
Vice President

MRS/csm

Enclosure

cc: NRC Clinton Licensing Project Manager
NRC Resident Office, V-690
Regional Administrator, Region III, USNRC
Illinois Department of Nuclear Safety
INPO Records Center

IE22
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NRC FORM 366 (4-95)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Loose Terminal Connection Causes Reactor Recirculation Pumps to Trip From Fast to Slow; Operation in the Restricted Zone, and Manual Scram

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 NAME	DOCKET NUMBER
									None	05000
06	13	96	96	008	00	07	09	96	FACILITY NAME: None	DOCKET NUMBER: 05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)	1								
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)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in 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 (Include Area Code)
D. K. Forbes, Maintenance Project Specialist	(217) 935-8881, Extension 3577

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 15 single-spaced typewritten lines) (16)

On June 13, 1996, the plant was in Mode 1 at about 100% reactor power. Control and instrumentation technicians were performing a calibration of the feedwater flow measurement instrument loop as part of a preventive maintenance task. The control and instrumentation technician was required by the preventive maintenance task to determine the voltage on a termination on a trip unit. When the technician attempted to take the voltage the two wire leads connected to the terminal separated, interrupting power to two trip units, which caused the reactor recirculation pumps to shift from fast to slow speed. The resulting reduction in core flow caused the plant to enter the restricted zone of the reactor thermal power versus core flow operating map. The exact cause of the loose wire could not be determined. Contributing to this event was the fact that it was difficult to attach the electrical meter to the point where it was required to take the voltage reading. Corrective actions for this event include installing special test lugs where needed, and a briefing with electrical and control and instrumentation technicians on this event.

DESCRIPTION OF EVENT

On June 13, 1996, at about 1719 hours, the plant was in Mode 1 (Power Operation) and about 100 percent reactor [RCT] power. Utility control and instrumentation technicians were performing preventive maintenance task PCIFWM135 which calibrates the feedwater [SJ] flow measurement instrumentation loop. This preventive maintenance task is performed every six months. As part of the preventive maintenance task, the control and instrumentation technician was required to determine the voltage on terminal 9 on the back of trip unit 1C34-K618B. When the control and instrumentation technician attempted to check the voltage, the two wire leads connected to this terminal separated, causing a loss of power to trip units 1C34-K626A and 1C34-K626B. This loss of power to the trip units created a false level 3 reactor water level signal and caused the reactor recirculation [AD] pumps [P] to shift from fast to slow speed. The shift of the reactor recirculation pumps from fast to slow speed caused a significant reduction in core flow.

At about 1720 hours, reactor power had stabilized at about 42 percent and core flow was about 28 million pounds per hour. The reactor operator, a licensed operator, recognized that the plant was now operating in the restricted zone of the reactor thermal power versus core flow operating map. The reactor operator immediately initiated a manual scram by placing the reactor mode switch [HS] in the shutdown position as required by procedure number 4008.01, "Abnormal Reactor Coolant Flow." As reactor water level decreased to the low reactor water level (Level 3) trip setpoint, containment isolation valves [ISV] in Groups 2 (Residual Heat Removal [BO]) to upper containment pools), 3 (Residual Heat Removal shutdown cooling), and 20 (miscellaneous) automatically closed or were already closed as designed. When reactor water level began increasing, the operators manually tripped the "A" turbine [TRB] driven reactor feedwater pump. Reactor water level recovered reaching the high water level (Level 8) trip setpoint which automatically tripped the "B" turbine driven reactor feedwater pump and the main turbine. The main turbine bypass valves [PCV] opened to control reactor pressure.

At about 1740 hours the reactor scram signal was reset. The plant was stabilized in Mode 3 (Hot Shutdown). Condition report 1-96-06-037 was initiated to investigate this event.

Following the event, it was apparent that the actions that the control and instrumentation technicians were performing initiated the shift of the reactor recirculation pumps from fast to slow speed. A review of the impact of losing power to the trip units confirmed that this would cause a shift of the reactor recirculation pumps from fast to slow speed. Also, investigation revealed that the termination that the control and instrumentation technician was trying to determine voltage on was slightly loose. There were no problems identified with the material condition of the terminal block, terminal screw or system wiring.

No other automatic of manually initiated safety system responses were necessary to place the plant in a safe and stable condition. No other equipment or components were inoperable at the start of this event to the extent that their inoperable condition contributed to this event.

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

CAUSE OF THE EVENT

The shift of the reactor recirculation pumps from fast to slow speed was caused by a loose connection on terminal 9 on the back of trip unit 1C34-K618B. This looseness allowed the wires terminated there to separate and interrupt current flow when the control and instrumentation technician attempted to determine voltage at that terminal point. The exact cause of the loose termination could not be determined. Contributing to this event was the difficulty the technician had in attaching the electrical meter to the point where it was required to take the voltage reading on the trip unit. A review of history at Clinton Power Station (CPS) revealed that the actuation of plant equipment due to maintenance technicians attaching electrical test equipment to terminal points on installed plant equipment has occurred infrequently at CPS.

CORRECTIVE ACTION

CPS will evaluate the Technical Specification surveillance procedures and preventive maintenance tasks for systems that could significantly impact plant operation and determine which ones require the attachment of electrical test and monitoring equipment on electrical terminal points. These terminal points will then be evaluated to determine which ones could adversely affect plant operation. The electrical terminal points that are identified as having the potential to adversely impact plant operation will be prioritized and scheduled to have special lugs installed at the terminal point to allow easy installation of temporary electrical test equipment. These lugs will be installed as plant conditions allow. The special lug has already been installed on Terminal 9 on the back of trip unit 1C34-K618B. Also, a briefing will be given to electrical and control and instrumentation technicians on this event. Included in this briefing will be information on the cause of the downshift of the reactor recirculation pumps, the contributing factors to this event and the corrective actions for the event.

ANALYSIS OF THIS EVENT

This event is reportable under the provisions of 10CFR50.73(a)(2)(iv) due to a manual actuation of the Reactor Protection System.

Assessment of the safety consequences and implications of this event indicated that this event was not nuclear safety significant. This event was analyzed and found to be consistent with the analyses of Decrease in Reactor Coolant Flow Rate transients found in Chapter 15 of the Updated Safety Analysis Report. This event was found to be within the design basis of the plant. The capability of the plant to perform its intended safety functions and achieve and maintain a safe shutdown was not affected by this event.

ADDITIONAL INFORMATION

Clinton Power Station has not reported other manual scrams having similar causes in recent history.

For further information regarding this event contact, Doug Forbes, Maintenance Project Specialist, at (217) 935-8881, extension 3577.