

ILLINOIS POWER

CLINTON POWER STATION, P.O. BOX 678, CLINTON, ILLINOIS 61727-0678, TELEPHONE (217) 935-8881

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10CFR50.73

Docket No. 50-461

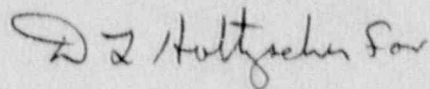
U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Clinton Power Station - Unit 1
Licensee Event Report No. 90-012-00

Dear Sir:

Please find enclosed Licensee Event Report No. 90-012-00: Normal End-of-Life Failure of Feedwater (FW) Flow Channel Power Converter Results in Sensed Low FW Flow, Recirculation Pump Down-Shift and Manual Scram. This report is being submitted in accordance with the requirements of 10CFR50.73.

Sincerely yours,



F. A. Spangenberg, III
Manager - Licensing and Safety

RSF/alh

Enclosure

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

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Clinton Power Station	DOCKET NUMBER (2) 0 5 0 0 0 4 6 1	PAGE (3) 1 OF 0 4
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TITLE (4)
Normal End-of-Life Failure of Feedwater (FW) Flow Channel Power Converter Results in Sensed Low FW Flow, Recirculation Pump Down-Shift 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 NAMES			DOCKET NUMBER(S)
0	5	1	7	9	0	9	0	0	None			0 5 0 0 0
0	5	1	7	9	0	9	0	0				0 5 0 0 0

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) 0 4 3	20.402(b)		20.406(c)	<input checked="" type="checkbox"/>	50.73(a)(2)(iv)		73.71(b)			
	20.406(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(iv)		73.71(c)			
	20.406(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(v)		OTHER (Specify in Abstract below and in Text NRC Form 366A)			
	20.406(a)(1)(iii)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(A)					
	20.406(a)(1)(iv)		50.73(a)(2)(iii)		50.73(a)(2)(viii)(B)					
	20.406(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(ix)					

LICENSEE CONTACT FOR THIS LER (12)									
NAME P. D. Yocum, Director-Plant Operations, extension 3205							TELEPHONE NUMBER 2 1 7 3 3 5 - 8 8 8 1		

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)									
CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC TURER	REPORTABLE TO NPRDS
E	J B C N V		X 9 9 9	Y					
X	A D R L Y		A 1 0 9	Y					

SUPPLEMENTAL REPORT EXPECTED (14)				YES (If yes, complete EXPECTED SUBMISSION DATE)	<input checked="" type="checkbox"/> NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
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ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single space typewritten lines) (16)

On May 17, 1990, with the plant in POWER OPERATION at 43 percent reactor power, and both reactor recirculation (RR) pumps in fast speed, reactor feedwater system (FW) flow channel "B" failed and caused instrumentation to incorrectly sense total FW flow as low. As a result of this, the logic initiated an automatic transfer of the "A" and "B" RR pumps to slow speed. In response to the RR pump transfers, the control room operator initiated a manual reactor scram in accordance with the off-normal procedure. Additionally, groups 2, 3 and 20 containment isolation valves actuated as a reactor vessel water low-level trip occurred. Troubleshooting determined that a power converter for the FW control system failed due to normal end of life and caused the "B" FW flow channel to fail. The cause of this event is attributed to the normal end-of-life failure of the power converter. Corrective actions included: replacing the specific power converter and replacing another power converter of the same model; initiating a Preventive Maintenance (PM) task to periodically replace the power converter and another converter of the same model; and determining if other FW control system power supplies/converters may need PM tasks for periodic replacement.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Clinton Power Station	DOCKET NUMBER (2) 0 5 0 0 0 4 6 1 9 0 - 0 1 2 - 0 0	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
					0 2	C 0 4

TEXT (If more space is required, use additional NRC Form 306A's) (17)

DESCRIPTION OF EVENT

On May 17, 1990, the plant was in Mode 1 (POWER OPERATION) and power ascension was in progress. The "A" and "B" Reactor Recirculation (RR) system [AD] pumps [P] were in fast speed.

At approximately 1631 hours, with the reactor [RCT] at forty-three percent of Rated Thermal Power (RTP), on the seventy-five percent flow control line of the power-flow map, Reactor Feedwater (FW) system [SJ] flow channel "B" failed and caused instrumentation to incorrectly sense total feedwater flow as low. As a result of this sensed low flow, the fifteen-second time delay relays [RLY] initiated the logic for automatically transferring the "A" and "B" RR pumps to slow speed.

At 1632 hours, the "A" and "B" RR pumps began the automatic transfer sequence to slow speed. The breaker [BKR] for the "B" RR pump opened first and that pump started to coast down in speed. The breaker for the "A" RR pump opened after the "B" pump coasted to below ninety-five percent of rated speed. (The cause for the delay in the "A" pump breaker opening is discussed in the CORRECTIVE ACTION section of this LER.) The difference in the opening times of the "A" and "B" RR pump breakers caused the fast to slow speed transfer logic to be bypassed for the "B" RR pump. Therefore, the "B" RR pump did not transfer to slow speed but tripped off-line. However, the "A" RR pump transferred to slow speed. This sequence resulted in reactor power decreasing to approximately twenty-one percent of RTP.

Because the "A" control room operator understood that both RR pumps should have transferred to slow speed, at 1635 hours, he placed the reactor mode switch [HS] in shutdown position and initiated a manual reactor scram in accordance with off-normal procedure 4008.01, "Loss of Reactor Coolant Flow."

Additionally, due to a normal level transient that immediately followed the reactor scram, a Reactor Vessel Water Low-Level (Level 3) trip occurred and resulted in activation of Groups 2 (Residual Heat Removal system [BO] to upper containment pools), 3 (Residual Heat Removal shutdown cooling) and 20 (miscellaneous) containment isolation valves [ISV].

At approximately 1636 hours, operators tripped the Main Turbine [TRB], and the Main Generator [GEN] immediately tripped on reverse power.

At approximately 1700 hours, operators completed the off-normal procedure checklist 4001.02C001, "Automatic Isolation Checklist," and verified that the Groups 2, 3 and 20 isolations had occurred as designed.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Clinton Power Station	DOCKET NUMBER (2) 0 5 0 0 0 4 6 1 9 0	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
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TEXT (if more space is required, use additional NRC Form 306A's) (17)

The plant was stabilized by 1809 hours in Mode 3 (HOT SHUTDOWN) with main steam isolation valves closed to minimize reactor cool-down. Reactor Pressure Vessel [RPV] level control was established using the Motor [MO] Driven Reactor Feed Pump.

Maintenance Work Request (MWR) D08903 was initiated to troubleshoot/repair the "B" FW system flow channel power converter. Another MWR, D13603, was initiated to troubleshoot/repair the timing sequence between the breakers for the "A" and "B" RR pumps.

No other automatic or 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 such that their inoperable condition contributed to this event.

CAUSE OF EVENT

Investigation determined that the power converter [CNV], 1C34-CNA, for feedwater control system [JB] signal resistor unit 1C34-SRU06 failed and caused the "B" feedwater flow, "C" reactor water level, and upset range water level channels to fail. Power converter 1C34-CNA had been energized for approximately nine years at the time of its failure. This failure was discussed with the power converter vendor, Arnold Magnetics, who indicated that failure at nine years of energization is considered a normal end-of-life failure for this converter. Since Illinois Power (IP) had no information relative to the normal life of this converter, a mechanism was not in place to periodically replace the converter. Additionally, since this power converter is not safety-related, it is not required to be in the scope of the equipment qualification program at Clinton Power Station (CPS).

CORRECTIVE ACTION

Power converter 1C34-CNA for signal resistor unit 1C34-SRU06 was replaced in accordance with MWR D08903. Additionally, only one other power converter at CPS is the same model as 1C34-CNA. This power converter, 1C34-CNB, was replaced in accordance with MWR D07529.

Preventive Maintenance (PM) tasks PCIFWA010 and PCIFWA011 have been established to replace the FW control system power converters 1C34-CNA and 1C34-CNB at five-year intervals. Arnold Magnetics has concurred with this interval.

IP will determine if other FW control system power supplies/converters in use may need PM tasks for periodic replacement. This determination is scheduled to be completed by September 28, 1990.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Clinton Power Station	DOCKET NUMBER (2) 0 5 0 0 0 4 6 1 9 0	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		0 1 2	0 0	0 0	0 4	OF	0 4

TEXT IF more space is required, use additional NRC Form 366A (17)

The cause for the "B" RR pump not transferring from fast to slow speed is attributed to RR logic time delay relays not actuating together. The control system for the RR pumps uses two fifteen-second time delay relays (one each for "A" and "B" RR logic) to input low feedwater flow conditions to the RR logic. A 0.5 second difference between the elapsed time of both time delay relays combined with RR pump coast-down characteristics can result in one of the RR pumps not transferring to slow speed as desired.

To correct the actuation sequence, time delay relay 1B33-K127A was adjusted in accordance with MWR D13603. Additionally, IP will investigate the RR pump logic for transferring to slow speed and will determine if changes to time delay settings or other design modifications are necessary to prevent future failures to transfer both RR pumps to slow speed. This investigation is scheduled to be completed by December 30, 1990.

ANALYSIS OF EVENT

This event is reportable under the provisions of 10CFR50.73(a)(2)(iv) because of the manual initiation of the reactor protection system [JC] and the activation of Groups 2, 3, and 20 containment isolation valves.

Assessment of the safety consequences and implications of this event has determined that this event was not nuclear safety significant. Prompt and correct operator action was taken by initiating a manual scram of the reactor in accordance with off-normal procedure 4008.01. Initiation of the reactor scram placed the plant in a safe condition.

ADDITIONAL INFORMATION

The power supply that failed during this event is model number SHU-24E(95-130) manufactured by Arnold Magnetics.

The time delay relays that were not actuating together during this event are model number 7012X10DL manufactured by the Agastat Relay Company.

No reportable events, with a similar cause, have occurred at Clinton Power Station.

For further information regarding this event, contact P. D. Yocum, Director - Plant Operations, at (217)935-8881, extension 3205.