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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. 96-002-00

Dear Sir:

Enclosed is Licensee Event Report No. 96-002-00: Spurious High Reactor Water Level Trip Signal Causes High Pressure Core Spray System to be Inoperable and Could Have Presented Fulfillment of the Safety Function. This report is being submitted in accordance with the requirements of 10CFR50.73.

Sincerely yours,

Wilfred Connell

Wilfred Connell
Vice President

RSF/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

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

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.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)
Spurious High Reactor Water Level Trip Signal Causes High Pressure Core Spray System to be Inoperable and Could Have Prevented Fulfillment of the Safety Function

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
01	31	96	96	002	00	02	27	96	FACILITY NAME: None	DOCKET NUMBER: 05000

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
	1		20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)			20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	100		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)	X	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)
K. C. Scott, Plant Engineering engineer	(217) 935-8881, Extension 3957

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)

With the plant in POWER OPERATION at about 100 percent reactor power, operators discovered the high pressure core spray (HPCS) system high reactor vessel water level seal-in and ready to reset lamps illuminated for no apparent reason. The lamps indicate that a high reactor vessel water level (Level 8) signal had been present for greater than 20 milliseconds and subsequently cleared, however, no Level 8 alarm was received. The Level 8 signal must be manually reset to restore the logic. With the signal not reset, the HPCS injection valve will not automatically open for all required conditions, specifically high drywell pressure and manual initiation. Therefore, the HPCS system would not have fulfilled its safety function of automatically initiating and injecting into the reactor vessel if required for a high drywell pressure condition while the Level 8 trip logic was not reset. The cause of this event is attributed to a spurious signal of indeterminate origin, most likely the result of an intermittent failure of a latch circuit component on the HPCS-1 logic circuit card. Corrective actions for this event include replacing the logic card and performing a bench test to determine the failure mode.

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

DESCRIPTION OF EVENT

On January 31, 1996, the plant was in Mode 1 (POWER OPERATIONS), at about 100 percent reactor [RCT] power. At about 1315 hours, during normal operator rounds of control room panels [PL], a control room operator discovered the high pressure core spray (HPCS) system [BG] high reactor vessel water level seal-in and ready to reset lamps [IL] illuminated for no apparent reason. Operators immediately checked reactor vessel water level instrumentation and verified that the level was satisfactory.

When illuminated, the high reactor vessel water level seal-in and ready to reset lamps indicate that a high reactor water level (Level 8) trip signal had been present from 2 transmitters [LT] for greater than 20 milliseconds and subsequently cleared. A Level 8 annunciator [ALM] should alarm when a Level 8 signal is initiated from either transmitter, however, no alarm occurred during this event. No maintenance or surveillance activities that would cause a Level 8 signal were in progress at the time of the discovery.

As indicated by the illuminated ready to reset lamp, the Level 8 signal must be manually reset to restore the logic. With the Level 8 signal not reset, the HPCS motor [MO]-operated injection valve [INV] will not automatically open for all required conditions, specifically high drywell pressure and manual initiation. Therefore, the HPCS system was declared inoperable. The length of time the lamps had been illuminated is not known; however, a thorough panel walkdown was performed between 0730 and 0745 hours on January 31, 1996, and the lamps were not observed to be illuminated at that time. Additionally, these lamps are very noticeable when illuminated and were probably illuminated for only a short period of time before they were noticed at about 1315 hours.

The Level 8 logic was reset at about 1330 hours, restoring the HPCS system to an operable status. Condition report 1-96-02-001 was initiated to track a cause analysis and corrective action determination for this event.

The Operations shift supervisor reviewed the event and concluded that the HPCS system would not have fulfilled its safety function of automatically initiating and injecting into the reactor vessel if required for a high drywell pressure condition while the Level 8 trip logic was not reset.

The HPCS system is designed to automatically start on a high drywell pressure signal or low reactor water level signal. The HPCS injection valve opens automatically on a HPCS initiation signal and closes automatically on a high reactor water level, Level 8, signal. If the water level drops to the low low reactor water level, Level 2, the valve will re-open and continue to cycle between Level 8 and Level 2 to maintain reactor core coverage with no operator action. This indicates the HPCS system would have responded appropriately to a valid low low water level signal during this event. Once the high water level signal has been initiated, it seals in and must be manually reset by depressing the high water level reset pushbutton [HS] before the valve can be operated manually. Additionally, with the Level 8 signal not reset, the HPCS injection valve will not automatically open on an automatic initiation of HPCS in response to high drywell pressure.

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

No 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 to the extent that their inoperable condition contributed to this event.

CAUSE OF EVENT

The HPCS system initiation logic circuits are automatically monitored and tested by a self-test system. In response to this event, technicians manually initiated the SYS-HPCS-1 and SYS-HPCS-2 self tests. The SYS-HPCS-1 test inputs into the HPCS-1 logic circuit card and monitors down stream logic for proper operation. The SYS-HPCS-2 test inputs into logic cards upstream of the HPCS-1 logic circuit card and monitors signals received by the HPCS-1 card. Neither of these tests identified any problems, and the seal-in and ready to reset lamps did not illuminate during the tests.

The cause of this event is attributed to a spurious high reactor vessel water level signal of indeterminate origin. The spurious signal was most likely the result of an intermittent failure of a latch circuit component on the HPCS-1 logic circuit card. The latch circuits are intended to allow initiation signals to seal-in when required and to prevent seal-in of signals resulting from self-test system pulses or electrical noise.

CORRECTIVE ACTION

The HPCS-1 logic circuit card will be replaced in accordance with maintenance work request (MWR) D50791 and bench testing of the card will be performed to determine the failure mode.

ANALYSIS OF EVENT

This event is reportable under the provisions of 10CFR50.73(a)(2)(v) because the Level 8 signal not being reset was a condition that alone could have prevented fulfillment of the HPCS safety function for high drywell pressure.

Analysis of the safety consequences and implications of this event identified that this event was not nuclear safety significant. With a high reactor vessel water level, Level 8, signal sealed in, the HPCS injection valve would not have opened on either a high drywell pressure signal or a manual initiation. Thus, the HPCS system would not have injected into the reactor vessel without operator action if called upon to do so in these instances. However, a low low reactor vessel water level, Level 2, signal would have acted as a Level 8 signal reset and allowed the system to inject into the vessel as required.

The Updated Safety Analysis Report (USAR) analysis lists the operational sequence of Emergency Core Cooling System (ECCS) for the Design Basis Accident (DBA), Loss of Coolant Accident (LOCA) large break reactor recirculation system rupture with a loss of offsite power. The analysis shows that a high drywell pressure signal is assumed to initiate at approximately time zero when the event occurs. About 3 seconds later, a Level 2 signal

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initiates. When the Level 2 signal initiates, the Level 8 signal seal-in would be reset. The HPCS injection valve does not start to open until the emergency diesel generator [EK] energizes the bus twelve seconds later. Therefore, the time delay of 3 seconds between the receipt of a Level 8 signal and a Level 2 signal is inconsequential in this accident scenario since power to open the injection valve is not available until 12 seconds into the event.

The USAR analysis for a small reactor recirculation system line break assumes that the HPCS system fails to perform its safety function. Thus, the illuminated seal-in and ready to reset lamps would not have limited the response to the small line break accident scenario.

In the event that the HPCS system would have been called upon to be manually initiated, it could be reasonably assumed that the reactor operator would observe the illuminated lamps and reset the Level 8 signal seal-in during the initiation process. This is a reasonable assumption since the Level 8 signal seal-in indication is in close proximity to the manual initiation pushbutton.

In addition, a probabilistic risk assessment (PRA) was modeled for the failure of the HPCS to initiate on a high drywell pressure while maintaining the ability of the HPCS to initiate on a Level 2 signal. The results of the PRA showed that under these conditions, no increase in core damage probability was predicted.

The HPCS system was found inoperable at about 1315 hours on January 31, 1996, and was restored to an operable status at about 1330 hours on January 31, 1996. The length of time the HPCS system was inoperable is unknown. The HPCS system is a single train safety system and has no redundancy.

ADDITIONAL INFORMATION

No equipment or components failed during this event.

Clinton Power Station has not reported similar events in recent history.

For further information regarding this event, contact K. C. Scott, Plant Engineering engineer, at (217) 935-8881, extension 3957.