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U-603305
2C.220

December 20, 1999

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. 1999-012-00

Dear Madam or Sir:

Enclosed is Licensee Event Report (LER) 1999-012-00: Failure of Original Design Supplier to Provide Sufficient Breaker Trip Setting Margin Results in Potential for High Pressure Core Spray Minimum Flow Valve to Fail Open and Less Than Design Flow to Reactor During Loss of Coolant Accident. This report is being submitted in accordance with the requirements of 10CFR50.73.

Sincerely yours,

Michael T. Coyle
Michael T. Coyle
Vice President

RSF/krk

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

PDR ADOCK 05000461

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

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TITLE (4)
Failure of Original Design Supplier to Provide Sufficient Breaker Trip Setting Margin Results in Potential for High Pressure Core Spray Minimum Flow Valve to Fail Open and Less Than Design Flow to Reactor During Loss of Coolant Accident

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
10	26	1999	1999 -	012 -	00	12	20	1999	None	05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100

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

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	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
	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)
M. G. McMenamin, Nuclear Station Engineering Department	(217) 935-8881, Extension 3469

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
B	BG	52	G082	Y					

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)

Operators were performing a High Pressure Core Spray (HPCS) system valve operability test and the system was inoperable. While stroking the HPCS to Suppression Pool minimum flow valve from the closed position to the open position, operators noted the Main Control Room indicator lights for the valve de-energized and a HPCS power loss status light energized. Local indication showed that the valve stayed in the open position. Operators manually isolated the affected containment penetration and shut the Reactor Core Isolation Cooling (RCIC) System storage tank suction valve to prevent draining the RCIC storage tank. Investigation determined that the breaker tripped during reversal of valve movement from the opening to the closing stroke due to a large inrush current. During a Loss of Coolant Accident the loss of power to the minimum flow valve could result in the failure of the HPCS minimum flow valve to close, providing less than design injection flow to the reactor. The cause of this event is attributed to the failure of the design supplier to provide sufficient margin in the trip setting of the breaker feeding the valve to prevent trips of the breaker from high inrush currents during reversal of valve movement. Corrective actions taken include changing the design trip setting of the breaker feeding the valve and reviewing other potentially affected circuit breakers for similar issues.

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

DESCRIPTION OF EVENT

On October 26, 1999, the plant was in Mode 1 (POWER OPERATION) at about 100 percent reactor [RCT] power and a Division 3 outage was in progress. The High Pressure Core Spray (HPCS) [BG] system was inoperable. At about 1819 hours, operators began performing surveillance procedure CPS 9051.02, "HPCS Valve Operability Test."

At about 1920 hours, while stroking the HPCS to Suppression Pool minimum flow valve [ISV] 1E22-F012 (a containment isolation valve) from the closed position to the open position in accordance with the surveillance procedure, operators noted the Main Control Room (MCR) indicator lights [IL] for the valve de-energized and the HPCS power loss or overload status light energized. Local indications showed that the valve stayed in the open position.

An area operator discovered the circuit breaker [52] for the valve had tripped open, de-energizing the valve. At 1920 hours, operators shut the Reactor Core Isolation Cooling (RCIC) System [BN] storage tank [TK] HPCS suction valve [V] to prevent the RCIC storage tank from draining into the Suppression Pool. At 1946 hours, in accordance with the Required Actions of Technical Specification 3.6.1.3, "Primary Containment Isolation Valves," operators manually closed valve 1E22-F012 to isolate the affected containment penetration [PEN]. The operator that closed motor-operated valve 1E22-F012 noted that the valve operated smoothly and exhibited no binding while closing.

Condition Report 1-99-10-177 was initiated to track a cause and corrective action determination for the valve failure. At about 2151 hours, operators reported the failure of the valve to the Nuclear Regulatory Commission (NRC) in accordance with 10CFR50.72(b)(2)(iii)(D) as a component failure that could have prevented an independent safety system from performing its function to mitigate the consequences of an accident. The loss of power to the HPCS minimum flow valve could have resulted in the failure of the HPCS minimum flow valve to close during a Loss of Coolant Accident (LOCA). In this scenario, the HPCS system would provide less than the HPCS design injection flow of 5,010 gallons per minute (gpm) to the reactor.

An investigation using recorders to monitor the breaker trip sequence and current determined that the breaker trip occurred during the reversal of valve movement from the opening stroke to the closing stroke. The design of the valve's control circuit causes the valve to close automatically when the valve is fully open, if the HPCS pump [P] is not running. During the valve test, the valve completed its opening stroke, so when the valve limit switches sensed the full open condition, the control logic reversed the valve's motor [MO] and caused the valve to start closing. The immediate reversal of the valve created a condition in which the forward momentum and inertia of the motor was being countered by application of reverse power. This condition caused an inrush current large enough to trip the breaker, de-energizing the control transformer [XFMR] in the Motor Control Center. With no power, the valve stopped closing and the MCR indicator lights for the valve de-energized.

The HPCS minimum flow valve breaker that tripped was installed in February 1999. The previous HPCS minimum flow valve breaker was installed in April of 1992, and had operated during surveillances with no reported tripping. In 1992, the valve breaker tripped during testing. An investigation at that time identified that the trip occurred at the point of reversal from the opening to the closing stroke, but was unable to quantify the actual magnitude of the transient current during the reversal; the breaker was replaced with a like-for-like replacement and no change to the design.

An engineering evaluation of this condition was completed on November 24, 1999. The evaluation concluded that the HPCS system would still perform its intended design basis safety function to prevent the peak fuel cladding temperature from exceeding the 10CFR50.46 limit of 2200 degrees Fahrenheit during the LOCA with the minimum flow valve in the open position. The engineering evaluation used the General Electric Clinton Power Station (CPS)-specific SAFER/GESTR Sensitivity Study (NEDC-31945P) as a basis for its conclusion. On the basis of this engineering evaluation, the HPCS system would still be capable of performing its safety function, so the NRC Notification was retracted on November 24, 1999.

On November 30, 1999, CPS recognized that the General Electric CPS-specific SAFER/GESTR Sensitivity Study utilized a methodology that is not currently established or approved in the CPS Licensing Basis for evaluating Emergency Core Cooling System (ECCS) cooling performance. Upon recognizing this information, CPS reassessed the event and determined it was reportable under the provisions of 10CFR50.72(b)(1)(ii)(B) as a condition during operation that resulted in the plant being outside its design basis. Condition Report 1-99-12-011 was initiated to document that this event was not reported within 30 days of the discovery date (October 26, 1999) as required by the provisions of 10CFR50.73.

CPS further concluded that the potential for the minimum flow valve to fail to close resulting in less than 5,010 gpm injection into the reactor was a condition prohibited by Technical Specification (TS) 3.5.1, "ECCS-Operating." TS 3.5.1 requires the HPCS system to be operable in Modes 1, 2 (STARTUP) and 3 (HOT SHUTDOWN).

No automatic or manually initiated safety system responses were necessary to place the plant in a safe and stable condition. Other inoperable equipment or components did not directly affect this event.

CAUSE OF EVENT

The cause of this event is attributed to the failure of the original design supplier to provide sufficient margin in the trip setting of the breaker feeding the 1E22-F012 valve to prevent trips of the breaker from high inrush currents during reversal of valve movement.

CORRECTIVE ACTION

Engineering Change Notice 31897 was issued to revise the size of the HPCS minimum flow valve breaker and the setpoint to a value high enough to preclude unintended tripping on inrush currents. The existing HPCS minimum flow valve breaker was replaced with a breaker having the revised setpoint.

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TEXT CONTINUATION

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

Other potentially affected circuit breakers were reviewed and no additional breakers are adversely affected by the failure mechanism discussed in this report.

ANALYSIS OF EVENT

This event is reportable under the provisions of 10CFR50.73(a)(2)(ii)(B) because it created a condition that was not in accordance with the design basis of the plant. This event is also reportable under the provisions of 10CFR50.73(a)(2)(i)(B) as a condition prohibited by Technical Specification 3.5.1.

The HPCS minimum flow valve breaker trip setpoint was not at a value high enough to preclude unintended tripping on inrush currents. This insufficient breaker design introduced the potential for the minimum flow valve to remain open during a HPCS injection and divert flow through the minimum flow line. In this situation, the design basis HPCS injection flow to the reactor assumed in the ECCS analysis (the design basis) would not be met. The unacceptable circuit breaker design has been in place since initial plant operation; however, the circuit breaker tripped only twice, once in 1992, and the second time in October of 1999, as discussed in this report.

An engineering evaluation of the safety consequences and implications of this event using the General Electric CPS-specific SAFER/GESTR Sensitivity Study (NEDC-31945P) concluded that the HPCS system would still perform its intended safety function to prevent the peak fuel cladding temperature from exceeding the 10CFR50.46 limit of 2200 degrees Fahrenheit with the minimum flow valve in the open position during the accident.

ADDITIONAL INFORMATION

The circuit breaker that failed during this event was an adjustable magnetic trip, molded case, Model TEC circuit breaker manufactured by General Electric.

Clinton Power Station has not reported similar events involving the design of circuit breaker trip settings.

For further information regarding this event, contact M. G. McMenamin, at (217) 935-8881, extension 3469.