NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	1 OF 0 4

TITLE (4)
Unit 2 Reactor Trip Following A Card Failure In The Process Control System

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 3	1 9	8 4	8 4	0 0 9	0 0	0 4	1 8	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 0 0

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

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Phillip B. Nardoci, Licensing Engineer	AREA CODE 7 0 4 3 7 3 - 7 4 3 2

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
X	J C	I M O D	W 1 2 0	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 fifteen single-space typewritten lines) (16)

On March 19, 1984 at 1501.41, a reactor trip was initiated by a Steam Generator A lo-lo level. This was caused by system fluctuations after a loop power supply (NLP) card failed in Channel 1 for the steam pressure signal to the 7300 Process Control System (PCS) which affected feedwater (CF) flow on Steam Generator (S/G) B. The failure caused the B S/G CF Control Valve to close. Operators placed the valve in "MANUAL", and then attempted to place the steam flow signal input for S/G B to Channel 2. S/Gs A, C, and D level controls were placed in "MANUAL." The feedwater pumps (FWPs) were put in "MANUAL". After the system was stabilized, FWP A was returned to "AUTOMATIC". This caused system transients (S/G levels began decreasing and S/G feedwater flows began fluctuating) which resulted in a reactor trip on S/G A lo-lo level. Unit 2 was in Mode 1 at 100% power at the time of the reactor trip.

The reactor tripped as designed and no system abnormalities resulted from this trip. The failed NLP card was replaced, the "Main Steam Line Pressure Calibration" procedure performed on S/G B, and the loop returned to service. This report (concerning the loss of a controlling steam flow channel) will be included in Operator Requalification Training.

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On March 19, 1984 at 1501:41, a reactor trip (and subsequent turbine trip) was initiated by a Steam Generator A [EIIS:GEN] lo-lo level. This was caused by system fluctuations after a loop power supply (NLP) card [EIIS:IMOD] failed in Channel 1 for the steam pressure signal to the 7300 Process Control System (PCS) [EIIS:JC] which affected feedwater (CF) [EIIS:SJ] flow on Steam Generator (S/G) B. The failure caused the B S/G CF Control Valve [EIIS:V] to close. Operators placed the valve in "MANUAL", and then attempted to place the steam flow signal input for S/G B to Channel 2. S/Gs A, C, and D level controls were placed in "MANUAL." The feedwater pumps (FWPs) [EIIS:P] were put in "MANUAL". After the system was stabilized, FWP A was returned to "AUTOMATIC". This caused system transients (S/G levels began decreasing and S/G feedwater flows began fluctuating) which resulted in a reactor trip on S/G A lo-lo level. Unit 2 was in Mode 1 at 100% power at the time of the reactor trip.

The purpose of the S/G level control [EIIS:JB] is to maintain a programmed water level in the S/Gs as a function of reactor power. Each S/G level valve controller consists of an independent control loop (with two independent channels per loop). Each loop contains a three element (level, steam flow, feedwater flow) feedwater valve controller. This controller compares actual feedwater flow to the steam flow and balances them when the level is correct. Control is accomplished by varying feedwater flow through the adjustment of the feedwater control valve to each S/G.

S/G water level is maintained using feedwater control valves. The feedwater control valve position is proportional to main feed pump speed which is varied to allow the feedwater control valves to be opened to the optimum control position (~60%). The pump speed is controlled according to the desired differential pressure (D/P) between the S/G inlet header and the steam header. The D/P is programmed to increase with increasing power. Totalized steam flow is used to derive the desired D/P program setpoint. The desired D/P program setpoint is compared to the actual D/P to provide a speed demand signal to the FWPs.

There are two feedwater flow channels for each S/G. The channel used for level control is selected using the S/G CF Flow switch (one switch per loop). The differency between feedwater flow and steam flow along with S/G level goes to the flow controllers. The feedwater control valve position is maintained proportional to the output of the flow controller.

There are two steam flow and pressure (compensated for flow) channels per S/G. The channel used for level control and feedwater pump speed control is selected using the "S/G STM FLOW SELECT" switch (1 switch per loop). The steam flow signal for the channel selected is supplied to the feedwater pump speed controller.

Manual or automatic operation of the FWP speed control is selected from the two manual/auto (M/A) stations. It is possible to operate either FWP in "MANUAL" speed control while the other is in "AUTOMATIC" control. In this case the flow balanced compensation will be defeated and the programmed D/P will be maintained automatically. If both M/A stations are in "MANUAL' control, FWP speed is controlled entirely by the Control Operator from the M/A stations. When both FWPs are in "AUTOMATIC", flow balance is maintained at the desired proportion. Total compensated steam flow from all four S/Gs provide the programmed D/P setpoint. A choice of two compensated steam flow channels (1 or 2) is available for each S/G through the use of the "S/G STM FLOW SELECT" switches. The selected

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

channels for control are added in the steam flow summing amplifier which produces the D/P program setpoint. This signal is sent to the Speed Demand Controller [EIIS:XC] which provides a pump speed demand signal for FWP speed control.

When the NLP card in the 7300 PCS failed, steam flow Channel 1 and steam pressure Channel 1 on S/G B indicated "ZERO". With Channel 1 selected for control, the B S/G CF Control Valve began closing, causing a decrease in feedwater flow to S/G B. Control Operators put the valve into "MANUAL" and reopened it to reestablish feedwater flow. The Control Operator stated that he then switched the "2B S/G STM FLOW SELECT" switch to Channel 2. He noticed the levels in S/G A, C, and D were fluctuating so he placed their level controllers in "MANUAL". He then put both FWPs into "MANUAL". Adjustments were made to the system to stabilize S/G levels and feedwater flow. All S/G levels were approximately equal at 1459 hours. FWP A was placed back into "AUTOMATIC". All four S/G levels began decreasing at the same rate as before the FWPs were put into "MANUAL". Control Operators made adjustments to S/G levels (S/G level control was still in "MANUAL"). When adjustments were made to increase the level in S/G C and S/G D, the levels in S/G A and S/G B decreased. S/G A reached the lo-lo level reactor trip setpoint which resulted in a reactor trip.

The steam flow and feedwater flow for the controlling channel in each loop are shown on the associated strip chart recorder [EIIS:XI]. For example, if the "S/G STM FLOW SELECT" switch is in the Channel 1 position for control and Channel 1 fails, the recorder will indicate the same information as the associated indicators. When the "S/G STM FLOW SELECT" switch is placed into Channel 2 for control, the green and red recorder pins should indicate the same value as seen on the Channel 2 indicators. If the recorder pins still indicate the Channel 1 value, this means that control is still aligned to Channel 1. The FWPs must be put in "MANUAL" because the FWP speed demand signal will be decreased, resulting in a decrease in FWP speed. The FWPs should not be returned to "AUTOMATIC" until the faulty channel is repaired or until the alternate channel is selected by placing the "S/G STM FLOW SELECT" switch to the alternate channel. The Control Operator stated that he placed the "2B S/G STM FLOW SELECT" switch to Channel 2 but he did not check the green and red pins on the recorder for a change. The recorder strip chart indicated that Channel 2 was not selected or that it failed to swap. If the relay card used when control is swapped from one channel to another (this is not the same card that failed in the PCS) was bad or if a fuse was open, control would have remained in Channel 1 even if Channel 2 was selected. Investigation found the card functioning properly.

The failed NLP card was replaced with a spare, the "Main Steam Line Pressure Calibration" procedure was performed which verified satisfactory operation of the control circuit, and the loop was returned to service. This report (concerning the loss of a controlling steam flow channel) will be included in Operator Requalification Training.

The reactor tripped as designed when the lo-lo level trip setpoint was reached on S/G A. This trip protects the reactor from the loss of heat sink in the event of a sustained steam/feedwater flow mismatch of sufficient magnitude to cause a serious drop in S/G level. No system abnormalities resulted from this trip.

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

Reactivity was properly controlled by the reactor trip. Pressurizer pressure dropped quickly following the trip, reaching a minimum pressure of 1980 psig. Pressure recovered smoothly to its reference value (2235 psig) within 30 minutes after the trip. The pressurizer PORV's and Code Safety Valves were not challenged, and pressure remained above the Safety Injection setpoint. Reactor coolant average temperature decreased quickly following the trip, and settled out to within ±2°F of Tno-load (557°F) about twelve minutes after the trip. The minimum temperature post-trip was 552°F. Pressurizer level responded smoothly to the trip, stabilizing near its no-load setpoint (25%) less than five minutes after the trip. The minimum level was 23%.

Steam pressure peaked at ~1136 psig. Pressure remained well below the setpoint for the first Main Steam Safety Valve bank (1170 psig.) Pressure stabilized at ~1085 psig.

Steam generator level was properly controlled, and level had recovered above its no-load target value (38%) within 10 minutes after the trip. Main feedwater was isolated as expected following the trip on reactor trip with coincident low average coolant temperature shortly after the trip. Auxiliary feedwater [EIIS:BA] initiated on low-low steam generator level and was used to recover and maintain level. This response was as expected.

Engineered Safeguards were not actuated. Normal power was available throughout this event. The health and safety of the public were unaffected by this incident.

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

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April 18, 1984

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

Subject: McGuire Nuclear Station, Unit 2
Docket No. 50-370
LER 370/84-09

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 370/84-09 concerning a reactor trip following a card failure in the process control system which is submitted is accordance with §50.73(a)(2)(iv). Initial notification of this event was made (pursuant to §50.72 Section (b)(2)(ii)) with the NRC Operations Center via the ENS on March 19, 1984. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

H.B. Tucker / BD

Hal B. Tucker

PBN:glb

Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

Records Center
Institute of Nuclear Power Operations
1100 Circle 75 Parkway, Suite 1500
Atlanta, Georgia 30339

Mr. W. T. Orders
NRC Resident Inspector
McGuire Nuclear Station

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