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 3

TITLE (4): Main Power Supply Failed in Process Control System Protection Cabinet I

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 4	2 3	8 4	8 4	0 1 1	0 0	0 5	2 3	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 (AREA CODE)
Phillip B. Nardoci, Licensing Engineer	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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	J C	J X	X 9 9 9	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On April 23, 1984 at approximately 0057, Control Room annunciator "Process Control System Power Supply Failure protection cabinet I" alarmed and status alarms indicated a loss of channel 1, which was being used for steam generator and pressurizer control. All four steam generator (S/G) levels began increasing because the four feedwater regulator valves opened. The Control Operators placed the four S/G level controls into "MANUAL". One Control Operator made adjustments with the level controllers to decrease levels in S/Gs A and B. Another Control Operator was ensuring that the controls were changed from channel 1 to channel 2. Then he began to make level adjustments on S/Gs C and D. The adjustments on S/G C and D were not made in time, which allowed S/G D to reach the hi-hi level trip setpoint of 82% at 0100. Unit 2 was in Mode 1, at 100% power, at the time of the turbine/reactor trip.

This event is attributed to Component Failure due to the failure of the main power supply in the Process Control System Protection Cabinet I. Design Deficiency also contributed to the event because power to both the main and back-up power supplies in each Protection Cabinet is supplied by one supply breaker for each cabinet.

The defective power supply was replaced with a spare, and operating procedures were revised. Modifications will be made to place the back up power supplies on separate supply breakers. The reactor tripped due to a turbine trip above 48% power. No system anomalies resulted from this trip.

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

On April 23, 1984 at approximately 0057, Control Room annunciator [EIIS:ANN] "Process Control System Power Supply Failure Protection Cabinet I" alarmed and status alarms indicated a loss of channel 1, which was being used for steam generator and pressurizer control. All four steam generator (S/G) [EIIS;GEN] levels began increasing because the four feedwater regulator valves [EIIS:V] opened. The Control Operators placed the four S/G level controls [EIIS:JB] into "MANUAL". One Contorl Operator made adjustments with the level controllers [EIIS:XC] to decrease levels in S/Gs A and B. Another Control Operator was ensuring that the controls were changed from channel 1 to channel 2. Then he began to make level adjustments on S/Gs C and D. The adjustments on S/G C and D were not made in time, which allowed S/G D to reach the hi-hi level turbine trip setpoint of 82% at 0100. Unit 2 was in Mode 1, at 100% power, at the time of the turbine/reactor trip.

This event is attributed to Component Failure due to the failure of the main power supply [EIIS:JX] in the Process Control System (PCS) [EIIS:JC] Protection Cabinet I. Design Deficiency also contributed to the event because power to both the main and back-up power supplies in each Protection Cabinet is supplied by one supply breaker [EIIS:BRK] for each cabinet.

Each Protection Cabinet is powered by a main 26.0 volt power supply (P/S). A back-up 24.0 volt power supply automatically provides power to the cabinet in the event the main power supply fails. Both power supplies in each protection cabinet are supplied by the same supply breaker, but the power supplies in each control cabinet are supplied by different supply breakers. Each power supply has a 35 ampere breaker and a 30 ampere fuse [EIIS:BRK] on the input with a 70 ampere breaker on the output. The Unit 2 protection cabinet I supply breaker is rated at 20 amperes.

Troubleshooting determined the main power supply was drawing excessive current (26.8 amperes) which tripped the 20 ampere supply breaker, but was not enough to open the 30 ampere fuse or open the 35 ampere breaker in the cabinet. The supply breaker trip resulted in a loss of both power supplies which caused a loss of protection channel 1. The system transients resulted in a S/G D hi-hi level trip two minutes after the power supply failed. The defective main power supply (North Electric Part No. PEC 3569) was removed from Protection Cabinet I and a spare was installed. It was satisfactorily tested and placed back in service. Modifications will be implemented on Units 1 and 2 to place the back-up power supplies for protection cabinets I, II, III, and IV on separate supply breakers. This will prevent deenergizing both power supplies if one supply breaker trips.

Unit 1 and 2 operating procedures were changed to include additional immediate action by operators when the "PCS Pwr Supply Failure Prot. Cab" alarm is received in the Control Room.

This was the fifth power supply failure in the PCS during the last 5 years (out of 32 power supplies on both units). The failures were caused by an open in the secondary winding of a transformer. This failure was the first resulting in a reactor trip. This was the first failure where the power supply did not disconnect itself by opening the 30 ampere fuse in the cabinet or by an open in the secondary of the transformer. The number of power supply failures is being investigated by Westinghouse to determine the cause and if other Westinghouse plants are experiencing similar power supply failures.

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

The turbine and feedwater pumps [EIIS:P] tripped as designed when the S/G hi-hi level trip setpoint was reached. This trip provides protection against excessive moisture carryover to the turbine [EIIS:TRB]. The reactor tripped due to a turbine trip above 48% reactor power. No system anomalies resulted from this trip. Reactivity was properly controlled by the reactor trip. Primary wide range pressure responded normally to the trip, reaching a minimum of 2006 psig before recovering. Pressure remained well below the PORV setpoint and well above the safety injection setpoint. The loop average temperature settled out at its expected value about 30 minutes after the trip. The average temperature did not drop below 557°F.

Main steam pressure peaked at 1109 psig. Pressure remained well below the steam generator PORV (1125 psig) and Main Steam Safety Valves setpoints (1170 psig.) Steam pressure was well controlled by the turbine bypass valves after the trip; remaining above ~1072 psig post-trip.

Following the trip, auxiliary feedwater initiated on loss of both main feedwater pumps. Minimum narrow range steam generator level was 25.7% which occurred about 5 minutes after the trip. Levels were well controlled and recovered smoothly to the post trip target value (38%) within 30 minutes after the trip.

No safety Injection actuation occurred. The pressurizer PORV's and code safety valves were not challenged. The primary temperature decrease was within the 100°F/hour Technical Specification Limit. The pressurizer level remianed on-scale. The S/G levels also remained on-scale. There was no abnormal NC leakage or release of radioactivity as a result of this event. The health and safety of the public were unaffected by this incident.

DUKE POWER COMPANY
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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
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May 23, 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-/84-11

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a)(1) and (d), attached is Licensee Event Report 370/84-11 concerning a reactor protection system actuation resulting from a failed main power supply in the process control system which is submitted in 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 April 23, 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 / PBN

Hal B. Tucker

PBN:glb
Attachment

cc: Mr. James P. O'Reilly
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