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 due to loss of both main feedwater pumps

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 0 0 0
0 4	1 9	8 4	8 4	— 0 1 0 —	0 0	0 5	2 1	8 4		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 J	I M O D	W 1 2 0	N					

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 19, 1984, the auxiliary AC electrical power system was being aligned in preparation to perform the "6.9 KV Normal Auxiliary Power Automatic Transfer Test." (The test was to be done on Unit 1.) In order to better control and protect the Unit 2 loads being supplied by Unit 1 sources, they were being transferred to Unit 2 sources. The transfers were done by the "dead bus" method, and operations personnel anticipated a slight decrease of feedwater (CF) pump 2B speed during the transfer. The transfer was initiated, and CF pump 2B speed began to decrease as expected. The transfer was completed approximately one second later, and CP pump 2B should have returned to speed; however, it did not. After approximately 38 seconds, the Operator attempted to trip the affected CF pump to initiate a 50% turbine/generator (T/G) runback. He inadvertently tripped CF pump 2A instead of CF pump 2B. Coincidently, the runback was not initiated (on one out of two CF pumps trip logic) because of a defective signal conditioning circuit card in the digital electro-hydraulic turbine control (DEH) cabinet. The turbine and reactor tripped at 1047, due to low steam generator (S/G) level.

This event is attributed to Component Failure/Malfunction due to the CF pump control oil orifice becoming blocked and the signal conditioning circuit card failing. Also contribuitng was Personnel Error, due to the operator inadvertently tripping the operating CF pump 2A. Unit 2 was at Mode 1 at 100% power at the time of this occurrence.

All 3 auxiliary feedwater pumps started to ensure that the reactor coolant system could be cooled. The CF pump control oil orifice was cleaned, and the circuit card replaced.

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

On April 19, 1984, the auxiliary A.C. electrical power system [EIIS:EF] was being aligned per procedure, "AC Electrical Operation Other Than Normal Line-up", in preparation to perform the "6.9 KV Normal Auxiliary Power Automatic Transfer Test". (The test was to be done on Unit 1.) The test verifies the operability of the automatic and manual transfer circuitry. The automatic transfer portion of this test submits a one second power lapse to some Unit 1 and shared equipment groups. Some of the Unit 2 shared loads were being supplied by Unit 1 sources at the time, and would suffer the one second loss of power. In order to better control and protect the Unit 2 loads, they were being transferred to Unit 2 sources. The transfers were done by the "dead bus" method, in which the existing power source is removed and then the other source is connected. (Power sources from the two units cannot be tied together.) Operations personnel anticipated a slight decrease of feedwater (CF) [EIIS:SJ] pump 2B [EIIS:P] speed during the transfer. The transfer was initiated, and CF pump 2B speed began to decrease as expected. The transfer was completed approximately one second later, and CF pump 2B should have returned to speed; however, it did not. The reason CF pump 2B did not return to speed was that the pump control oil (LP) [EIIS:LL] was blocked due to a clogged orifice. The foreign material that clogged the orifice apparently resulted from the transient, caused by the one second power lapse, which broke material loose from LP piping downstream of the LP filters [EIIS:FLT]. After approximately 38 seconds, the Operator attempted to trip the affected CF pump to initiate a 50% turbine/generator (T/G) runback. He inadvertently tripped CF pump 2A instead of CF pump 2B. Coincidently, the runback was not initiated (on one out of two CF pumps trip logic) because of a defective signal conditioning circuit card [EIIS:IMOD] in the digital electro-hydraulic turbine control (DEH) [EIIS:JJ] cabinet. The turbine and reactor tripped at 1047, due to low steam generator (S/G)[EIIS:GEN] level.

This event is attributed to Component Failure/Malfunction due to the LP oil orifice becoming blocked and the signal conditioning circuit card failing. Also contributing was Personnel Error, due to the operator inadvertently tripping the operating CF pump 2A. Unit 2 was at Mode 1 at 100% power at the time of this occurrence.

The shifting of the power supplies, by the "dead bus" method, was completed within one second. The "dead bus" transfer method is used when the two shared load center sources are out of synchronism. After a time delay, of approximately one second, the residual voltage on the bus has decayed to an acceptable level and the stand-by breaker [EIIS:BRK] will close. CF pump 2B controller [EIIS:XC] sustained one second power lapse due to the transfer. This one second transient decreased CF pump 2B speed, and caused foreign material to break loose and block the governor valve servomotor orifice. The operator mistakenly tripped the operating CF pump 2A, 38 seconds later.

Unit 2 reactor tripped due to a low-low level in 2C S/G in anticipation of a loss of the normal heat sink for the reactor coolant system [EIIS:AB]. All three auxiliary feedwater pumps started to ensure that the reactor coolant system could be cooled down to less than 350°F. (Two S/Gs low-low level conditions.)

When the unit is above 50% load the failure of a 50% capacity CF pump, will result in the DEH system running back the electrical load to 50%, to prevent the unit from a trip. When CF pump 2B was inadvertently tripped, a runback should have occurred. It was later determined that the signal conditioning circuit card had failed, due to a diode shorting. When the diode shorted, it rendered DEH runback signal inoperable. Therefore, if the correct CF pump had been tripped in a timely manner, the unit would have tripped because the runback circuit was inoperable and the unit could not have continued to operate on CF pump 2A alone.

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

The CF pump 2B control oil orifice was cleaned, and the DEH signal conditioning circuit card was replaced.

The 6.9 KV switchgear normal to stand-by source automatic transfer test is required every 18 months. The automatic transfer will transfer the source of incoming power from the normal to the alternate auxiliary transformer. This could be accomplished by either a rapid or slow bus transfer. The rapid bus transfer will occur if the two transformer power supplies are initially in synchronism (determined automatically in the transfer circuit by the position of the breaker auxiliary contacts) and the associated reactor coolant pump [EIIS:P] is operating. When synchronism is not verified by circuit breaker position, a slow transfer occurs. The rapid transfer occurs within approximately eight cycles and the slow occurs within approximately one second. The test is presently written to be performed in Mode 5, as a slow transfer. Pre-operational testing of the rapid transfer scheme successfully transferred incoming power sources without loss of auxiliary loads. The "6.9 KV Normal Auxiliary Power Automatic Transfer Test" will be rewritten to include the ability of satisfying the acceptance criteria for the automatic transfer by the rapid transfer when the reactor is subcritical and the reactor coolant pumps are operating.

Reactivity was properly controlled by the reactor trip. Pressurizer pressure reached a minimum of 1995 psig before recovering to its reference value (2235 psig). Pressurizer level reached a minimum of 21.6% before returning to normal post-trip value of 25%. Average coolant temperature reached a minimum of 549.6°F before recovering to its normal post-trip value of 557°F. This minimum is a few degrees lower than normal; it was caused by the extended auxiliary feedwater flows used to recover S/G levels following the trip. The temperature response was within the technical specification cooldown limits.

Steam pressure peaked at ~1125 psig. One steam generator PORV opened for 46 seconds. (The setpoint is 1125 psig.) Steam pressure dropped to ~1010 psig following the trip as a result of the extended injection of auxiliary feedwater at a high rate. Pressure recovered to ~1040 when auxiliary feedwater was throttled, and increased to ~1060 when flow was further reduced. Pressure remained well below the main steam safety valve setpoints. Following the trip, auxiliary feedwater initiated on low-low steam generator level. All 3 pumps started on low-low steam generator level and supplied the steam generators. Main feedwater was isolated as expected on reactor trip with coincident low primary average temperature.

Narrow Range steam generator levels were offscale low for 4.5 minutes after the trip. This is not expected. Auxiliary feedwater was being supplied at full flow rate at this time. Levels in all four generators were near or at the low-low level setpoint at the time of the trip. When the voids collapsed post-trip, level dropped offscale. Level did not recover until sufficient mass was added by the auxiliary feedwater system. However, primary to secondary heat transfer was maintained at all times, as Tcold and Psteam trend together. Auxiliary feedwater flow was reduced to 200 gpm per generator once the levels had recovered above the post-trip low-low level setpoint. Steam generator levels had recovered within 5% of their expected post-trip values about thirty minutes after the trip. No safety injection actuation occurred. The pressurizer PORVs and code safety valves were not challenged. Indicated pressurizer level remained on scale. The average coolant temperature decrease was within the 100°F/hour Technical Specification Limit. There was no abnormal release of radioactivity during this event, and no abnormal coolant leakage. Although steam generator narrow range levels were off-

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

scale low for 4.5 minutes, primary-to-secondary heat transfer was maintained at all times. The health and safety of the public were not affected by this incident.

DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

May 21, 1984

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

Subject: McGuire Nuclear Station, Unit 2
Docket No. 50-370
LER 370/84-10

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 370/84-10 concerning a Unit 2 Reactor Trip due to the loss of both main feedwater pumps 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 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 / TBO

Hal. B. Tucker

PBN:glb
Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
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NRC Resident Inspector
McGuire Nuclear Station

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