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 1	0 5 0 0 0 3 6 9	1 OF 9

TITLE (4): Reactor Trip on Main Feedwater Pump Trip

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 1	2 8	8 5	8 5	— 0 0 4 —	0 0	0 2	2 7	8 5		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
Scott Gewehr - Licensing	AREA CODE 7 0 4 3 7 3 - 7 5 8 1

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)

[ ] 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 January 28, 1985, McGuire Unit 1 tripped from 100 percent power on Lo-Lo Steam Generator Level when a main feedwater pump tripped on low suction pressure. The cause of the pump trip was the failure of a pneumatic pressure transmitter, which provided a false low-pressure signal. Following the pump trip, a turbine runback did not occur due to the failure of a component in the turbine control system. The failure of the turbine to run back caused the Lo-Lo level trip.

Two other post-trip abnormalities occurred during this event. A Steam Generator power-operated relief valve remained open for about four minutes, causing excessive cooldown and loss of Steam Generator inventory. Also, a discharge check valve on the turbine-driven auxiliary feedwater pump failed to close after the pump was secured, causing backflow from the direction of the Steam Generator to the pump and upper surge tank. Other anomalies are described in the body of the report.

The major cause of this event is considered to be component failure due to the failed pressure transmitter, which initiated the pump trip, and to the failure of the turbine control system to run back the turbine. Corrective actions will repair or replace failed components, as well as determine ways to preclude future failures, or to minimize their significance.

The health and safety of the public were unaffected by this incident.

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INTRODUCTION: On January 28, 1985, the Unit 1 reactor tripped on Steam Generator (S/G) "B" lo-lo level. The low S/G level condition was caused by the loss of the Main Feedwater Pump Turbine (FWPT) 1B which had tripped on low suction pressure. A low suction pressure did not actually exist but a pneumatic pressure transmitter had failed, giving a false low pressure signal to trip FWPT 1B.

A turbine runback signal following the loss of FWPT 1B was not initiated due to an electronic component failure in the Digital Electro Hydraulic (DEH) turbine control system. Without the turbine runback to 56%, all four S/G levels dropped to the lo-lo level Reactor Trip setpoint in approximately one minute. A manual runback was attempted, unsuccessfully.

Two significant post trip abnormalities occurred during this event. 1) The S/G "B" power operated relief valve (PORV) opened and remained open for approximately four minutes causing excessive cooldown and loss of S/G inventory. This failure was due to a pressure instrument calibration drift. 2) 1CA-49, discharge check valve on the turbine driven auxiliary feedwater pump (AFWPT), failed to close as the pump was secured following the reactor trip. This failure caused much of the water flow to S/G "C" to backflow into the upper surge tank and AFWPT, and not into the S/G, until the check valve was isolated.

Other post-trip abnormalities are listed in the Transient Analysis section of this report.

Unit 1 was in Mode 1 at 100 percent power at the time of this event.

A component failure is the cause of this event for the following reasons: 1) FWPT 1B suction pressure transmitter failure, and 2) DEH input circuit component failure. A contributing design deficiency, existed on the FWPT suction pressure instrumentation to the extent that one pneumatic pressure transmitter supplied the signal to three pressure switches for a 2 out of 3 (2/3) logic trip signal.

EVALUATION: On January 28, 1985, at 1024, the Unit 1 reactor tripped on S/G "B" lo-lo level. The low S/G level resulted from the loss of FWPT 1B without an automatic turbine runback actuation.

The unit transient began when a pneumatic pressure transducer on the suction line of the FWPT 1B failed. The failure was caused by a rubber air tube rupture inside the transmitter.

The pneumatic transmitter, 1CMPT5970, measures the FWPT 1B suction pressure and supplies a proportional instrument air signal to four pressure switches (1CMPS5970, 1CMPS5971, 1CMPS5972, 1CMPS5973) and a pneumatic indicator. Pressure switch 1CMPS5970 provides a low pressure alarm at 280 psi decreasing. Pressure switches 1CMPS5971, 1CMPS5972, and 1CMPS5973 provide a 2 out of 3 (2/3) logic FWPT 1B trip at 230 psi decreasing and an events recorder alarm. The pressure transmitter failure caused all the output pressure switches and the indicator to decrease, causing a feedwater pump trip on 3/3 logic. This situation of a single component failure initiating a multiple logic trip was identified on February 16, 1983 in a Nuclear Station Modification (NSM) request. This requested NSM identified deficiencies on both the suction and discharge instrumentation of all four McGuire FWPTs. This modification is being implemented on the Unit 2 FWPTs during the current Unit 2 1985 refueling outage. The Unit 1 modification is scheduled for completion during the 1985 refueling outage for that Unit. The modifications will remove two of the pressure switches from the transmitter output and place them on the actual process line.

The pneumatic transmitter is an Ashcroft series 4000 unit. The tubing failed on what is identified as the "thin walled" side of the tubing and appeared as a split. Station personnel do not believe that this tubing failure is a generic problem as no other cases have been identified.

The main turbine runback signal generated in this event originated from the FWPT 1B control oil low pressure switch. This oil pressure switch actuated immediately after the FWPT suction pressure transmitter failed low to begin the main turbine runback.

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

After being processed by electrical relay logic, the runback signal is fed to a 48 VDC signal conditioning circuit card and processed for input to the DEH automatic runback circuits. The automatic turbine runback did not occur because the diode on this circuit card (D1) had failed and was electrically shorted. This short circuit prevented relay K1 from energizing, preventing the runback signal from reaching the DEH circuitry.

The purpose of this diode in the circuit is to limit the inductive voltage surge following deenergizing of the direct current (DC) relay coil.

Nine similar diode failures have been found since the Unit 1 and Unit 2 DEH systems were placed in service. These failures were reported to Westinghouse on June 28, 1984. Westinghouse responded to this report with a letter giving recommendations to check the circuit card jumper configuration, determine if induced noise is present on the input lines and ensure that all the input cables are shielded.

It has been verified that all the input circuit card jumpers are installed correctly and that shielded cable is used on most of the input cables. The area where shielded cable was not used included the area termination cabinets (ATC) where field connections are made. The ATCs are normally prewired with unshielded cable prior to the cabinet installation. Electrical noise measurements were made with no abnormally high voltages found on any of the input cards.

A permanent resolution to the diode failure problem is being pursued. Plans include:

1) Determination if the diode is necessary for circuit operation and, if not, remove it.

2) Determination if a diode of higher rating is necessary for proper circuit operation.

3) Review of the possibility of providing redundant input circuits or a manual input circuit.

The main steam PORV actuation was caused by the main steam (SM) pressure transmitter, 1SMPT5510, calibration drift. This instrument is similar to the Ashcroft series 4000 pneumatic transmitter that is used on FWPT suction pressure. The calibration drift of this instrument is attributed to the lack of preventive maintenance (PM). Maintenance records show that this instrument has experienced calibration drift in the past. The scheduling priority for the PM on this instrument was low enough that it did not get placed on the maintenance schedule to be completed. The instrument has been repaired/calibrated only when a failure or drift was noticed. The drift on this type of instrument is normally due to either clogged or dirty instrument air orifices or mechanical wear. According to calibration data, the pressure transmitter had drifted high by approximately eight percent. This would have caused the PORV to close at approximately eight percent below the normal setpoint.

The pressure transmitter senses the SM pressure and converts it to a proportional instrument air signal. This output air signal operates the pressure switches which open and close the PORV. The deadband for the opening and closing duration of the PORV is controlled by the difference in setpoints between the two pressure switches.

The failure of auxiliary feedwater (CA) check valve, 1CA-49, was due to the failure of the valve clapper to close as the AFWPT was stopped. The failed open check valve allowed some of the discharge flow from the motor driven (M/D) AFWP to flow back toward the AFWPT and through the minimum flow recirculation valve (1CA-20) to the UST. The stop check valve (1CA-22) on the discharge of the AFWPT also did not close and allowed some of this backflow to enter the pump casing and suction piping. The suction relief valve (1CA-128) opened providing a flow path through the pump. This flow was stopped when the isolation valve (1CA-50) between the AFWPT and the S/G "C" was closed. The suction pressure instruments were overranged and declared inoperable per Technical Specification 3.3.2.7d.

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On January 29, 1985, at 0210, the electrical isolation valve used to isolate 1CA-49 was to be opened for normal valve alignment. The valve would not open using the motor actuator. A Nuclear Equipment Operator (NEO) was sent to open the valve manually. As the valve was manually opened, flow was heard. The valve was reclosed and it was determined that 1CA-49 was still leaking. The AFWPT suction instruments were overranged and declared inoperable again per Technical Specification 3.3.2.7d.

Maintenance personnel disassembled 1CA-49 (Borg-Warner swing check valve) and checked for internal damage and improper operation. No major damage was found. The seating surfaces were lapped and the valve reassembled. The check valve tested satisfactorily for backleakage and forward flow.

Apparently during this forward flow operational test, the check valve stuck open again. As the flow was being decreased, following this test, the suction pressure instruments began to increase. Control Room Operators quickly isolated the check valve before the suction pressure exceeded 60 psi and damaged the suction instruments again.

Maintenance personnel began disassembling the check valve again to check for damage. As the bonnet bolts were loosened, the check valve clapper was heard dropping back into position. At this point, the valve bonnet bolts were retorqued and the valve was checked for back leakage. No leakage was detected. Normal valve alignment was achieved without problem and the Unit 1 reactor startup resumed.

Similar valves on the Unit 2 auxiliary feedwater system have experienced two failures due to an internal seal weld on the valve seat breaking. The report which describes these events is LER 370/84-25. The check valves used on Unit 2 are also Borg-Warner components but are not the same as the Unit 1 check valves. The Unit 2 valves (Item no. 6J21) are rated for 1500 psi working pressure. Both models are swing check valves with bonnet mounted clappers. The Unit 2 check valve failures (2CA-49) on August 26 and 30, 1984 were also believed to be caused by a stuck open clapper in conjunction with a broken seal weld. The Unit 1 failure in this report is the first occurrence of a stuck open valve of this model at McGuire.

The manufacturer has been contacted for a solution to the sticking problem. The possibility of adding a mechanical stop to the clapper arm to prevent the clapper from traveling up to the valve body is the major plan for permanent repair of this valve.

PRE-TRIP TRANSIENT

Prior to the reactor trip reactor power was reduced ~5% by the operators during the runback. Pressurizer pressure peaked at 2266 psig at the the reactor trip. The pressurizer PORVs and code safety valves were not challenged. Reactor coolant loop average temperature increased approximately 1.5 °F because of the imbalance between heat production and removal. Reactor coolant loop differential temperature decreased as reactor power was reduced. Pressurizer level went up with average coolant temperature to a peak of 63%.

Steam pressure increased from ~1000 psig to ~1040 psig during the runback. Steam generator levels dropped from ~67% to ~56% at the time of the reactor trip because of the underfeed. With the slower manual runback and one operating main feedwater pump, the feed flow was not sufficient, and the unit tripped on low-low steam generator level.

POST-TRIP RESPONSE

Reactivity was properly controlled by the reactor trip. Immediately after the reactor trip pressurizer pressure dropped to ~2025 psig and continued to decrease with average coolant temperature. The pressure reached a minimum value of 1972 psig before recovering. This was well above the Safety Injection setpoint (1845 psig). Pressure stabilized at ~2140 psig fifteen minutes after the reactor trip, below the reference value of 2235 psig.

Average coolant temperature dropped to ~542 °F nine minutes after the trip because of the steam pressure reduction resulting from the steam generator B PORV closing ~100 psi below its nominal setpoint and extended auxiliary feedwater addition. The minimum average coolant temperature was 541.2 °F, below the expected no-load target of 557 °F. Temperature stabilized once the steam generator PORV closed and auxiliary feedwater flow was throttled. The cooldown rate was 48 °F per hour, within the Technical Specification limit of 100 °F per hour.

Pressurizer level dropped following the trip as steam pressure and average coolant temperature declined. Letdown isolated on low pressurizer level (17%). The minimum pressurizer level was 16.8%. A second charging pump was manually started and used to recover level. Letdown was reestablished about six minutes after it was isolated, and the second charging pump was secured about three minutes thereafter. Level had recovered to ~35% (no-load target is 25%) by that time. No change in reactor coolant pump status occurred during this event.

After the reactor trip, steam pressure reached a maximum of 1138 psig in steam generator A. The A and D steam generator PORVs remained closed, while the B and C steam generator PORVs opened. The steam generator C PORV lifted at a pressure of 1094 psig

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

decreasing, after the pressure had peaked. It closed at a pressure of 1084 psig. The B steam generator PORV opened at a pressure of 1123 psig and remained open for ~4 minutes. The valve did not close until steam pressure had fallen below 993 psig, 100 psig below the nominal reseat value. The PORV was isolated about two minutes after it closed. Reactor coolant cooldown and steam generator inventory loss occurred while the steam generator B PORV was open. Steam pressure continued to drop after the steam generator B PORV closed, because auxiliary feedwater was feeding the steam generators at a high rate. When the auxiliary feedwater flow was reduced, pressure settled out at ~950 psig. This was well below the no-load target of 1092 psig, but well above the main steam isolation setpoint of 585 psig. No steam generator main steam safety valves lifted during this event.

At the time of the reactor trip, all four steam generator levels were ~56%. As steam pressure spiked, the level in steam generators A, B and C dropped to ~24% before recovering. As steam pressure decreased, the steam generator B level reached ~8% narrow range before turning. Level then increased in steam generator B because of increased voiding while the pressure decreased. Then level began to decrease as the inventory loss continued. When the B steam generator PORV reseated, B steam generator level dropped off-scale on the narrow range indication, reaching a minimum of 42% wide range level. Narrow range level remained off-scale for ~6 minutes until it was restored with auxiliary feedwater. Auxiliary feedwater flow to S/Generators C and D was halted about nine minutes after the reactor trip, as level had recovered above the no-load target in these two generators. At that time the discharge of the Turbine Driven auxiliary feedwater pump was directed solely to the A and B steam generators. Level then quickly recovered in steam generator B. Flow was then adjusted as needed for steam generator level control. Thirty minutes after the trip the steam generator level were stable and were within 6% of the no-load target value of 38%.

SAFETY EVALUATION

1. The reactor was shutdown and reactivity was controlled by the insertion of the control rods on the reactor trip.

2. Residual heat was removed by auxiliary feedwater to the atmosphere and the condenser. Level in steam generator B was offscale low on the narrow range indication for about 6 minutes after the reactor trip. As the primary temperature in this loop trended with the others, primary to secondary heat transfer was maintained at all times.

3. No unusual release of radioactivity occurred during this event. Adequate core cooling was maintained at all times. The reactor coolant pressure boundary remained intact. Letdown was isolated on low pressurizer level, but the low level was caused by NC system shrinkage due to the cooldown, not as a result of leakage.

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CORRECTIVE ACTION: The FWPT 1B suction pressure transmitter has been repaired and recalibrated.

The DEH input circuit has been repaired and tested for automatic runback.

The S/G "B" steam line pressure transmitter and PORV setpoints have been recalibrated.

Maintenance personnel disassembled and inspected 1CA-49 for internal damage. The major probelem still existing is the fact that the check valve is binding internally and sticking in the open position. The valve can be closed by loosening the bonnet bolts and freeing the clapper but has not closed automatically in the last two attempts at normal usage.

NSMs will be completed during the 1985 refueling outages which will eliminate the possibility of a single component failure (pressure transmitter) causing a FWPT trip on low suction or high discharge pressure.

Possible corrective actions are being reviewed to eliminate the diode failures in the DEH input circuits.

The main steam transmitters which actuate the S/G PORVs will be placed on an active PM schedule to help detect calibration drift before it becomes excessive.

Borg-Warner has been contacted for recommendations for repair of the 1CA-49 check valve. The most desirable solution currently available is to place a mechanical stop on the clapper to prevent overtravel. NSMs have been written to replace both Unit 1 and Unit 2 Borg-Warner check valves on the auxiliary feedwater system. NSMs have also been written to replace the pump discharge stop check valves.

The health and safety of the public were not affected by this event.

DUKE POWER COMPANY
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HAL B. TUCKER
VICE PRESIDENT
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February 27, 1985

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

Subject: McGuire Nuclear Station, Unit 1
Docket No. 50-369
LER 369/85-04

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 369/85-04 concerning a Reactor Trip on a Feedwater Pump Trip and Consequent Low-Low Steam Generator Level which is submitted in accordance with §50.73 (a)(2)(iv). 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/BB

Hal B. Tucker

SAG/mjf

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator
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