Duke Power Company
McGuire Nuclear Generation Department
12700 Hagers Ferry Road (MG01A)
Huntersville, NC 28078-8985

T. C. McMeekin
Vice President
(704)875-4800
(704)875-4809 Fax

May 8, 1992

DUKE POWER

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

Subject: McGuire Nuclear Station Unit 2
Docket No. 50-370
Licensee Event Report 370/92-06

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 370/92-06 concerning a Unit 2 Reactor Trip. This report is being submitted in accordance with 10 CFR 50.73 (a) (2) (iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

T.C. McMeekin

T.C. McMeekin

TLP/bcb

Attachment

xc: Mr. S.D. Ebneter
Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, GA 30323

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, GA 30339

Mr. Tim Reed
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. P.K. Van Doorn
NRC Resident Inspector
McGuire Nuclear Station

IE22
1/1

LICENSEE EVENT REPORT (LER)

FACILITY NAME(1)	DOCKET NUMBER(2)	PAGE(3)
McGuire Nuclear Station, Unit 2	05000 370	1 OF 8

TITLE(4) Unit 2 Experienced a Turbine / Reactor Trip Due To Equipment Failure and an Unknown.

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)
									N/A	05000
04	09	92	92	06	0	05	08	92		05000

OPERATING MODE(9): 1

POWER LEVEL(10): 100 %

THIS REPORT IS SUBMITTED PURSUANT TO REQUIREMENTS OF 10CFR (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)
	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	AREA CODE	TELEPHONE NUMBER
Terry L. Pedersen, Manager, McGuire Safety Review Group	704	875-4487

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
F	KA	PT	I204	yes						

SUPPLEMENTAL REPORT EXPECTED(14)		EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	NO				

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

On April 9,1992, at 1303, Unit 2 experienced a transient which resulted in a Reactor Trip from 100 percent power. Pressure transmitter, 2CMPT5190, which controls valve 2CM-58, Condensate Cooler Bypass, failed low on a high differential pressure causing valve 2CM-58 to close. Condensate Booster Pumps A and C were in service and tripped on emergency low pump suction pressure. Hotwell Pump B automatically started. Condensate Feedwater Pump Turbines (CFPT) A and B tripped upon loss of all Condensate Booster Pumps. Generator Load Rejection Bypass Control Valve 2CM-420 did not adequately modulate as Condensate Booster Pump suction header pressure decreased. The loss of both CFPTs resulted in a Turbine Generator trip and subsequent Reactor trip. Units 1 and 2 were in Mode 1 (Power Operation) at 98 and 100 percent power, respectively, at the time of the event. Plant equipment responded as required and Operations personnel implemented procedure EP/2/A/5000/01, Reactor Trip or Safety Injection, and stabilized Unit 2 in Mode 3 (Hot Standby). This event is assigned causes of Equipment Failure and Unknown. The 4 hour notification was made to the NRC at 1518, on April 9, 1992, in accordance with procedure RP/0/A/5700/10, NRC Immediate Notification Requirements. The failed pressure transmitter was replaced.

EVALUATION:

Background

The Condensate (CM) system [EIIS:SD] takes condensate from the Condenser [EIIS:COND] Hotwell, purifies it to meet water chemistry specifications, heats it to improve the thermal cycle efficiency, and delivers it to the Main Feedwater (CF) system [EIIS:SJ] for makeup to the Steam Generators (SG) [EIIS:SG].

Differential pressure transmitter (DPT) [EIIS:PDT] 2CMPT5190 monitors Stator Cooling Water Heat Exchanger [EIIS:HX] differential pressure and sends a signal to the computer [EIIS:CPU]. An annunciator [EIIS:ANN] alarm [EIIS:ALM] is given on low pressure. During normal power operation, two Hotwell Pumps (HWPs) are in service, with the third HWP in standby. The standby HWP is started on low-low pressure signal which is also sent to the Operator Aid Computer (OAC) [EIIS:ID] and events recorder. The Condensate Booster Pumps (CBPs) [EIIS:P] are tripped when 2 of 3 pressure switches [EIIS:PS] indicate emergency low suction pressure following a 50 second time delay, and another signal is sent to the events recorder. A signal is also sent to bypass valve [EIIS:V] 2CM-420, Generator Load Rejection Bypass Control Valve, which during normal operation is closed, but will modulate to maintain CBP suction header pressure above the desired setpoint. To prevent Unit 2 from tripping during a load rejection, valve 2CM-420 must modulate open to allow added condensate supply flow to the CBPs.

The Main Feedwater (CF) system [EIIS:SJ] takes the treated CM system water, heats it further to improve the plant thermal cycle efficiency, and delivers it at the required flow rate, pressure, and temperature to the SGs for makeup.

The CF system uses 2 turbine [EIIS:TRB] driven feedwater pumps (CFPT) designated as A and B. Each CFPT will trip when 2 of 3 suction pressure switches detect a low-low suction pressure signal or loss of all CBP. At > 56 percent Turbine Generator [EIIS:TG] load, the loss of either CFPT will initiate a Turbine Generator runback signal. An automatic Main Turbine trip is initiated on the loss of both CFPTs.

If a Turbine trip occurs above approximately 48 percent Reactor [EIIS:RCT] power, the Reactor protection interlock, P-8 permissive, ensures that an automatic Reactor [EIIS:RCT] trip will be initiated.

Description of Event

On April 9, 1992, at approximately 1303, Operations (OPS) Control Room (CR) [EIIS:NA] personnel received an alarm from Annunciator 2AD-8, window D-6 indicating a low CBP suction header pressure. The Reactor Operator at the Controls (ROATC) referenced the OAC graphic display for the CM system. The OAC reflected a CBP suction header pressure of 52 psig and decreasing. CBPs A and C were in service at the time. A low-low suction header pressure alarm followed by an emergency low suction header pressure alarm was received. The ROATC verified that HWP B, had automatically started, and was running. He again referenced the OAC and noted a CBP inlet pressure of 27 psig. At 1303:58 CBPs A and C tripped on emergency low suction pressure. CBP B, which had been in standby, automatically started. The ROATC asked the Balance of Plant Operator (BOP) to verify that valve 2CM-420 was open. The ROATC noted that the status light for valve 2CM-420 indicated the valve was closed. The BOP reported that the manual loader for valve 2CM-420 was positioned in its normally full open position. The loss of the CBPs caused the tripping of CFPTs A and B, which in turn resulted in tripping the Turbine Generator. Because the Turbine tripped at > 48 percent full power, the P-8 interlock disengaged the Unit 2 Reactor Trip breakers at 1303.59. At 1304, CR personnel manually exercised the Reactor Trip breakers, implemented procedure EP/2/A/5000/01, Reactor Trip or Safety Injection, and stabilized the Unit in Mode 3 (Hot Standby).

Conclusion

This event is assigned a cause of Equipment Failure/Malfunction because DPT 2CMPT5190 failed causing a low pressure signal to be generated on a high differential pressure. The failure caused a "close" signal to be sent to valve 2CM-58, Condensate Cooler Bypass, which regulates the bypass flow around the Condensate, Generator Stator Water, and Hydrogen Coolers [EIIS:CLR]. As 2CM-58 closed, a low and emergency low CBP suction header pressure alarm was generated. HWP B, which had been in standby, automatically started as condensate flow to the CBP suction header decreased. After a 50 second time delay, CBPs A and C tripped on emergency low suction pressure. CFPTs A and B tripped upon the loss of CBPs A and C. When CFPTs A and B tripped and because the Turbine Generator was operating at > 56 percent load, a runback signal was generated. The runback caused 2CM-420 to open and CBP B automatically started. Additionally, the tripping of both CFPTs resulted in an automatic start of the Auxiliary Feedwater (CA) pumps and a Main Turbine Generator trip signal was generated. Because the Unit was operating at > 48 percent Reactor power when the Turbine Generator tripped, a Reactor trip was initiated by the P-8 interlock.

2CMPT5190 is an ITT Barton, Model 273A Differential Pressure Transmitter. Identification of the failed component was performed by OPS personnel who surmised the problem was between the HWP and CBP. Three suspect components are in this flow path: (a) valve 2CM-58, (b) the Gland Steam Condenser differential pressure valves, and (c) the Polishing Demineralizers. OPS personnel proceeded to valve 2CM-58 and requested CR personnel to close valve 2CM-420. A decrease in suction flow was noted. OPS personnel then opened valve 2CM-57, bypass around valve 2CM-58, and noted that the CBP suction header pressure increased. This indicated that valve 2CM-58 was closed. The failure was traced to DPT 2CMPT5190. Instrument and Electrical (IAE) personnel examined the component but were unable to determine the exact failure mechanism. During the initial post trip investigation, IAE personnel attempting to calibrate 2CMPT5190 noted that the transmitter would only accept a 12 pound signal before failing low. The normal operating signal range is 3-15 lbs. IAE technician A attempted to increase the output by pressing the flapper against the nozzle. The signal did not increase to 15 lbs. as expected. The technician then placed his finger over the nozzle and an immediate increase in pressure was noted. This would indicate that some portion of the instrument air that is normally deflected by the flapper to modulate valve 2CM-58 was escaping in an uncontrolled manner.

Conversations with the Component Engineering personnel and IAE technicians who performed the field examination of 2CMPT5190 indicated that a small "chip" immediately adjacent to the transmitter nozzle orifice was noted. It is theorized that this indication could have permitted sufficient air to leak by to cause the flapper portion of the transmitter to interpret this signal as an increase in differential pressure. Sensing an increase in differential pressure, 2CMPT5190 sent an "open" signal to valve 2CM-58 in an effort to direct less condensate through the Stator Coolers and decrease the differential pressure. The flapper continued to exert downward pressure on the nozzle, as a result of the leak by air, sending more control air to open valve 2CM-58 further. The downward pressure of the flapper reached a point that normally would equate to a full open valve position. When this position was passed, the instrument air coupled with the leak by air caused a deflection of the flapper. This deflection allowed an excessive amount of instrument air to escape from the nozzle. This was interpreted by the transmitter as an extremely low differential pressure across the Stator Coolers and a "close" signal was sent to valve 2CM-58. The immediate closing of valve 2CM-58, resulting from this signal, caused the diversion of all condensate flow to the Condensate, Stator, and Hydrogen Coolers, thus decreasing CBP suction header pressure. The decrease in CBP suction header pressure was the initiating event.

This scenario was tested at the McGuire Nuclear Technical Training Center on April 30, 1992, with an identical ITT Barton Transmitter. The test indicated that the transmitter would fail in a like manner under the conditions described above.

A cause of Unknown, Possible Inappropriate Action is also assigned to the event because the air filter regulator [EIIS:AFRG] for valve 2CM-420, Generator Load Rejection Bypass Control Valve, had no output control air. Valve 2CM-420 is controlled by electrical to pneumatic converter [EIIS:CNVR] 2CMEP4200 and pressure regulator [EIIS:RG] 2CMPR4200, to maintain the desired CBP suction pressure. Solenoid valve [EIIS:PCV] 2CMSV4200 is energized by a load rejection signal to allow the control of valve 2CM-420 by the manual loader in the CR. This allows the Unit to endure a transient during load rejection by maintaining adequate feedwater flow. Normally, as CBP suction header pressure decreases, valve 2CM-420 modulates to stabilize the header pressure. Subsequent post trip investigation noted that although the input air for the air regulator for valve 2CM-420 was normal, there was no indication of control air output. With no control air output, valve 2CM-420 could not respond to the decreasing CBP header pressure as designed. Upon

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME(1)	DOCKET NUMBER(2)	LER NUMBER(6)			PAGE(3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
McGuire Nuclear Station, Unit 2	05000 370	92	06	0	5	OF	8

receipt of the runback signal, generated by the CFPT trip, control of valve 2CM-420 was automatically shifted to the manual loader in the CR. The manual loader is maintained in the full open position. The "open" signal generated by the runback was sent to valve 2CM-420 through the manual loader in the CR and valve 2CM-420 responded correctly and began to supply condensate flow to the suction header.

This investigation examined the equipment histories of valve 2CM-420 and its associated components but found no evidence that the 0 percent control air output was related to any previous maintenance work performed on the controller or associated components. Discussions with IAE personnel performing the field investigation ruled out inadvertent positioning by personnel, or mispositioning as the result of system conditions, i.e. vibration. No apparent damage of the filter air regulator (2CMPR4200) that would account for the 0 percent air output was noted in the preliminary examination. The component was examined by the Metallurgy Laboratory for evidence of internal failure. No evidence of internal failure was noted.

A review of the Operating Experience Program data base for the 24 months prior to this event revealed five Reactor trips attributable to equipment failure. These events are detailed in Licensee Event Reports (LERs) as follows:

1) 370/92-04 - Unit 2 Tripped due to a failed valve positioner and possible installation deficiency.

2) 370/91-07, Manual Reactor/Turbine trip due to axial flux difference approaching Technical Specification limit with no ability to move rods.

3) 370/91-12, Unit 2 Reactor Tripped due to Shutdown Bank C dropping.

4) 369/90-27, Reactor/Turbine trip occurred on an SSPS general warning.

5) 370/91-01, Reactor Trip occurred due to a loss of offsite power.

Therefore, this problem is considered to be recurring. The corrective actions formulated for the LERs above were specific to the individual events and affected equipment and would not have precluded this event.

CORRECTIVE ACTIONS:

Immediate: 1) OPS CR personnel manually exercised the Reactor Trip breakers, implemented procedure EP/2/A/5000/01, and stabilized the Unit in Mode 3.

Subsequent: 1) Work Request 147286 was generated and DPT 2CMPT5190 was replaced.

2) Work Request 146895 was generated to investigate valve 2CM-420 to determine the cause of 0 output control air.

3) The air controller for valve 2CM-420 was examined by the Duke Power Metallurgy Laboratory to determine the possibility of internal failure.

4) DPT 2CMPT5190 was examined in an effort to determine the failure mechanism.

5) OPS personnel notified the NRC of the event at 1518 in accordance with procedure RP/0/A/5700/10, NRC Immediate Notification Requirements.

Planned: 1) OPS will identify balance of plant valves important to unit reliability and determine a desired failsafe position.

2) Component Engineering personnel will evaluate the feasibility of reconfiguration of control loops to achieve the failsafe position desired by OPS.

3) OPS will develop a periodic inspection of valve 2C -420 and its associated components to verify operability.

SAFETY ANALYSIS:

An analysis of transients and accidents postulated (e.g. Turbine Trip, Loss of Normal Feedwater Flow) which could result in a reduction of the capacity of the secondary system to remove heat generated in the Reactor Coolant System [EIIS:AB] is presented in Section 15.3, Decrease In Heat Removal By The Secondary System, of the Final Safety Analysis Report (FSAR). For a Turbine trip event, discussed in Section 15.2.3 of the FSAR, the turbine stop valves close rapidly on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals as described in Section 10.2.? of the FSAR. The automatic steam dump system would normally accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere and main feedwater flow would be lost. For this situation, feedwater flow would be maintained by the CA system to ensure adequate residual and decay heat removal capability. Should the steam dump system fail to operate, the steam generator safety valves may lift to provide pressure control.

Since 1970, the American Nuclear Society (ANS, classification of plant conditions has been used which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. A Turbine trip is classified as an ANS Condition II event, a fault of moderate frequency. A loss of normal feedwater is classified as an ANS Condition II event, a fault of moderate frequency, also.

The loss of normal feedwater analysis (Section 15.2.7.2 of the FSAR) is performed to demonstrate the adequacy of the Reactor protection and Engineered Safeguards System (e.g. CA) in removing long term decay heat and preventing excessive heatup of the NC system with possible resultant NC overpressurization or loss of coolant. Following a loss of normal feedwater, CA is capable of removing the stored and residual heat, thus preventing either overpressurization of the NC system or loss of water from the Reactor core, and returning the plant to a stabilized condition. Also, for a loss of normal feedwater, the steam dump to the condenser is assumed to be lost, heat removal from the secondary system then occurs through the steam generator power operated relief valves or safety valves. Since no fuel damage is postulated to occur, radiological consequences resulting for this transient would be less severe than the steamline break accident analyzed in Section 15.1.5.3 of the FSAR.

The Unit responded to the Reactor trip without any significant problems. All primary and secondary system parameters were at their approximate no load values 30 minutes after the trip.

The Main steam pressure did not reach the Main Steam Power Operated Relief Valve (PORV) or Main Steam Code Safety Valve lift setpoints and the valves were not challenged. The Reactor Coolant system pressure did not reach the Pressurizer [EIIS:PZR] PORV or Pressurizer Code Safety valve lift setpoints and the valves were not challenged. Adequate core cooling was maintained throughout this transient, and the Reactor Coolant system boundary was not challenged. Emergency power and emergency core cooling were not required in this event and were not actuated.

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

ADDITIONAL INFORMATION

Sequence of Events:

TR - Trip Report
LB - Logbooks (SRO, RO)
ER - Plant Events Recorder

DATE	Time	Event
4/9/1992	1303:58	Low-low and emergency low CBP suction header pressure alarm

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME(1)	DOCKET NUMBER(2)	LER NUMBER(6)			PAGE(3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
McGuire Nuclear Station, Unit 2	05000 370	92	06	0	8	OF	8

received.(ER,TR)

Standby HWP (B) automatically started. (ER,TR)

CBPs A and C tripped on low suction pressure (ER,TR)

CFPTs A and B tripped. (ER,TR)

Runback signal was generated. (ER)

Standby CBP (B) automatically started. (ER,TR)

1303:59.024 Turbine Generator tripped. (ER,TR)

1303:59.109 Unit 2 Reactor Trip breaker A opened. (ER,TR)

1303:59.120 Unit 2 Reactor Trip breaker B opened. (ER,TR)

1304:02 Motor Driven Auxiliary Feedwater Pump (MDAFWP) A started. (ER,TR)

1304:04 MDAFWP B started. (ER,TR)

1305:15 Turbine Driven Auxiliary Feedwater Pump started. (ER,TR)

1304:06.494 Unit 2 Reactor Trip breaker A manually exercised. (ER,TR)

1304:06.500 Unit 2 Reactor Trip breaker B manually exercised. (ER,TR)

1518 OPS personnel notified NRC in accordance with procedure RP/0/A/5700/10. (SRO).