Duke Power Company
McGuire Nuclear Station
P.O. Box 488
Cornelius, N.C. 28031-0488

(704) 875-4000

DUKE POWER

November 30, 1989

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 370/89-03-01

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 370/89-03-01 transmitting additional information as committed in LER 370/89-03 dated May 8, 1989. 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,

Tony L McConnell

T.L. McConnell

DVE/ADJ/cbl

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

M&M Nuclear Consultants
1221 Avenue of the Americas
New York, NY 10020

American Nuclear Insurers
c/o Dottie Sherman, ANI Library
The Exchange, Suit 245
270 Farmington Avenue
Farmington, CT 06032

Mr. Darl Hood
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

bxc: B.W. Bline
A.S. Daughtridge
J.S. Warren
R.L. Gill
R.M. Glover (CNS)
T.D. Curtis (ONS)
P.R. Herran
S.S. Kilborn (W)
R.E. Lopez-Ibanez
J.J. Maher
R.O. Sharpe (MNS)
G.B. Swindlehurst
K.D. Thomas
L.E. Weaver
R.L. Weber
J.D. Wylie (PSD)
J.W. Willis
QA Tech. Services NRC Coordinator (EC 12/55)
MC-815-04
(20)

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

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 6

TITLE (4) Reactor Trip Occurred Because Of The Failure Of The Positioner For Steam Generator 2C Main Feedwater Regulating Valve

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	0 6	8 9	8 9	0 0 3	0 1	0 5	0 8	8 9	N/A	0 5 0 0 0
										0 5 0 0 0

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

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

[ ] 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	
Alan Sipe, Chairman, McGuire Safety Review Group	7 0 4	8 7 5 - 4 1 8 3

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	S J	B E	M 4 3 0	Yes					

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 6, 1989 at 0503, a Unit 2 Reactor Trip occurred because of Low-Low Level in Steam Generator 2C. The unit was operating in Mode 1 (Power Operation) at 100% power prior to the trip. The Low-Low Level condition was caused by the failure of the bellows in the positioner for valve 2CF-20, Steam Generator 2C Main Feedwater (CF) Regulating Valve. The Turbine Generator automatically tripped because of the Reactor Trip. Operations personnel implemented the Reactor Trip recovery procedure to recover from the transient. At 0542, Operations personnel made the required notification to the NRC. Instrumentation and Electrical personnel completed the replacement of the failed feedback bellows in the positioner for valve 2CF-20 and the other 3 Unit 2 S/G CF flow control valves at 1459. The failed bellows was sent to the Duke Power Company Metallurgy Laboratory for analysis. The Refueling Outage Preventive Maintenance (PM) for the CF regulating valves will be revised to replace the feedback bellows assemblies on these valves. Unit 2 was returned to Power Operation on April 7, 1989 at 0123. This event is assigned a cause of Equipment Failure because of the failure of the feedback bellows in the positioner for valve 2CF-20.

EVALUATION:

Background

The Main Feedwater (CF) System [EIIS:SJ] includes a CF regulating valve [EIIS:FCV] and an isolation valve [EIIS:ISV] in the flow path to each Steam Generator (S/G) [EIIS:SG]. These valves are designed to fail to the closed position. If either valve fails closed during full power operation, the water level in the affected Steam Generator would begin to decrease rapidly. When operating at full power, there is no practical way to keep the affected S/G from reaching the Low-Low Level setpoint which would initiate an automatic Reactor [EIIS:RCT] Trip.

In a process control loop, the controller output represents the required position the actuator must assume to satisfy the process requirements. The actuator may not necessarily assume this required position because of packing stem friction, and high pressure drops across the valve plug. The valve positioner corrects for these conditions by comparing the actual position of the valve to the controller output signal. If the valve position does not coincide with the control signal, the positioner will continue to supply air or exhaust it until the actuator position on the control valve agrees with that called for by the controller. The positioner for valve 2CF-20 that failed was a Moore model No. 72P315.

Description of Event

On April 6, 1989, at 0500, while Unit 2 was operating at approximately 100% power with no major problems, an alarm was received in the Control Room indicating low feedwater flow to S/G 2C. The Control Room [EIIS:NA] Reactor Operator took manual control of valve 2CF-20, S/G 2C Main Feedwater Regulating Valve, in an attempt to open the valve. Subsequently, there was no increase in CF flow or level in S/G 2C. The output for valve 2CF-20 controller [EIIS:XC] indicated 100% output demand. The Senior Reactor Operator (SRO) instructed the Control Room Reactor Operator to commence reducing Turbine [EIIS:TRB] Generator [EIIS:GEN] load at a rate of 5 MW/min. initially, and when S/G level did not increase the SRO instructed the Control Room Reactor Operator to increase the load reduction rate to 20 MW/min. The level in S/G 2C continued to decrease and as level approached the trip setpoint, the Control Room Reactor Operator was instructed by the SRO to manually trip the Reactor at 41% S/G level. However, approximately 2 seconds prior to the Control Room Reactor Operator exercising the Reactor Trip Breakers [EIIS:BRK], the Reactor automatically tripped at 0503 on S/G 2C Low-Low level. The Turbine Generator automatically tripped because of the Reactor Trip. Motor Driven Auxiliary Feedwater System [EIIS:BA] Pumps [EIIS:P] 2A and 2B automatically started in response to S/G 2C Low-Low level. A CF system Isolation did not occur because Reactor Coolant (NC) system [EIIS:AB] T-ave stabilized above the CF system isolation setpoint (553 degrees-F). Valve 2CF-107, S/G 2D CF Bypass Control, did not control properly in automatic; consequently, the Control Room Reactor Operator took manual control of the valve.

Operations (OPS) Control Room personnel made the required notification to the NRC according to the NRC Immediate Notification Requirements procedure, RP/0/A/5700/10, at 0542.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	8 9	— 0 0 3 —	0 1	0 3 OF 0 6

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

OPS personnel initiated emergency priority Work Request 138239 at 0600, to have Instrumentation and Electrical (IAE) personnel investigate and repair valve 2CF-20. IAE personnel found that the feedback bellows in the positioner for valve 2CF-20 had failed. IAE personnel replaced the feedback bellows at ~1459, on April 6, 1989.

Unit 2 was returned to Power Operation on April 7, 1989, at 0123.

Conclusion

This event was assigned a cause of Equipment Failure because of the failure of the feedback bellows in the positioner for valve 2CF-20. Valve 2CF-20 failed in the closed position as designed. There are two possible reasons why the feedback bellows failed. One reason is that of valve vibration and another reason could be the manufacturing process employed by the manufacturer, Moore, in soldering the feedback bellows assemblies. After a number of bellows failures were attributed to the manufacturing process, Moore changed its manufacturing process sometime after 1984. IAE personnel and Moore personnel are investigating whether or not the failure that occurred was a part of the deficient manufacturing process.

IAE personnel previously initiated Problem Investigation Report (PIR) No. 0-M89-0079 on March 27, 1989 because of a control valve positioner failure on 2CF-23, S/G B CF Regulating Valve. As a result of this PIR, positioners for valves 1 and 2 CF-23, were replaced and the valve positioners on the other S/G CF regulating valves were inspected for air leakage. After the failure of valve 2CF-20, IAE personnel replaced the Unit 2 CF regulating valve bellows which were constructed in September, 1987 with the same type bellows; however, the replacement bellows were manufactured in November, 1987. IAE personnel have ordered more feedback bellows assemblies from Moore that were constructed after November, 1987 to replace existing assemblies and to ensure the reliability of the product. The Unit 1 CF regulating valve bellows assemblies have already been replaced with feedback bellows assemblies that were constructed after November, 1987. The Unit 2 CF regulating valve bellows assemblies will be replaced with the more recently constructed assemblies by the end of the 1989 Refueling Outage. The valve operators for the S/G CF regulating valves undergo preventive maintenance (PM) every 5 years and the control loop for these valves undergo PM every Refueling Outage. A potential failure of the bellows probably would not have been detected by these PMs because of the location of the failure. The failure could only be detected through destructive testing.

OPS Control Room personnel responded to the transient in a timely manner to stabilize the unit. There were a few abnormalities noted during this Reactor Trip. NC system Loop D Overpower Delta Temperature setpoint gave erratic indications after the trip. IAE personnel made adjustments in the loop and functionally verified the loop operable. Valve 2CF-107 would not control properly in automatic. IAE personnel replaced a process controller card, (NCB), and functionally verified the loop operable.

All primary and secondary system key parameters, with the exception of those noted above, responded as expected during this Reactor Trip. Approximately 30 minutes after the Reactor Trip, Pressurizer [EIIS:PZR] level and pressure had all achieved stable no-load conditions. S/G 2C narrow range level was trending upward. S/G no load pressure and NC system T-ave were slightly elevated because of the Steam Dump [EIIS:JI] controller being set to control at ~1100 psig.

A review of McGuire LERs for the previous 12 months revealed two other Reactor Trips that were attributed to an equipment failure; however, none of the equipment failures were caused by a failed bellows in a valve positioner. Therefore, according to the Nuclear Safety Assurance guidelines this event is considered not recurring. This is the second occurrence of a failed feedback bellows assembly in a valve positioner; therefore, failed feedback bellows assemblies in valve positioners is considered a recurring problem.

The Post Reactor Trip Plant Response is classified as a Category A since all transient classification criteria fell within Category A (plant responses remained within preferred or expected bounds).

This event is reportable to the Nuclear Plant Reliability Data System (NPRDS). A review of the NRPDS data base indicated that there have been a number of valve positioner failures. The failures were attributed to vibration, unknowns, air leaks, dirt, normal wear and aging.

There were no personnel injuries, radiation overexposures, or releases of radioactive material as a result of this incident.

CORRECTIVE ACTIONS:

Immediate: Operations personnel implemented the Reactor Trip recovery procedure, AP/2/A/5500/01.

Subsequent:

1) IAE personnel replaced the feedback bellows on the positioners for all four Unit 2 S/G CF flow control valves.

2) IAE personnel sent the failed feedback bellows for valve 2CF-20 and a feedback bellows assembly that was replaced but had not failed to the Duke Power Company Metallurgy Laboratory for analysis.

Planned:

1) IAE personnel will send the failed feedback bellows for valve 2CF-23 and a feedback bellows assembly that was replaced but had not failed to Moore Products for analysis.

2) IAE personnel will revise the Refueling Outage PM for the Unit 1 and 2 CF regulating valves to include replacing the feedback bellows assembly.

3) The McGuire Safety Review Group will submit a revised LER with additional information upon receipt of the failure analysis for the feedback bellows.

SAFETY ANALYSIS:

The Unit 2 Reactor tripped on S/G 2C Low-Low level. The Turbine Generator Trip was automatic as a result of the Reactor Trip. This Reactor Trip is bound by the "Loss of Normal Feedwater Flow" event of the McGuire Final Safety Analysis Report (FSAR), Chapter 15, Section 15.2.7. The event described in the FSAR is more limiting because it assumes a complete loss of CF flow. The CA system is assumed to provide decay heat removal capability following an automatic Reactor Trip from Low-Low S/G water level. The water level in S/G 2C did not go as low as predicted in the FSAR; therefore, the transient was less severe than that analyzed in the FSAR. The CA system started automatically as designed and provided necessary additional feedwater flow to all four S/Gs to assist in returning S/G water level to normal.

All primary and secondary system parameters necessary to ensure a safe shutdown were at or approaching no-load conditions approximately 30 minutes after the trip with the exception of S/G no load narrow range level which was trending upward. S/G no load pressure and NC system T-ave were slightly elevated because the Steam Dump controller was set to control at ~1100 psig. S/G Power Operated Relief Valves [EIIS:RV] (PORVs), S/G Code Safety Valves, NC system PORVs or Code Safety Valves were not challenged. This Reactor Trip presented no hazard to the integrity of the NC system or Main Steam systems [EIIS:SB]. There were no radiological consequences as a result of this event.

This event did not affect the health and safety of the public.

Additional information is provided as a revision to this LER based on the results of the failure analysis for the feedback bellows.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
McGuire Nuclear Station, Unit 2	0 5 0 0 0 3 7 0	8 9	0 0 3	0 1	0 6 OF 0 6

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

ADDITIONAL INFORMATION:

The following information addresses planned corrective action number 3, to submit information on the results of the failure analysis for the feedback bellows.

Prior to April 26, 1989, IAE personnel had sent the failed feedback bellows for valve 2CF-20 and a feedback bellows assembly that had been replaced but had not failed to the Duke Power Company Metallurgy Laboratory for analysis. The analysis results revealed that the Moore positioner inner bellows for valve 2CF-20 failed because of a circumferential fracture. The crack propagated approximately 120 degrees around the bellows circumference and showed no gross deformation. Destructive examination of the failed bellows showed the failure to be a circumferential tear. Destructive examination of the unfailed Moore positioner bellows assembly indicated the device contained no cracks or gross deformations. The bellows assemblies of the failed and unfailed Moore positioners were inspected for extraneous solder constraining the bellows and causing failures. No evidence was visible to indicate extraneous solder was associated with the current failures. The failure of the bellows is attributed to cyclic stresses that were induced by normal system vibrations.

On May 6, 1989, IAE personnel sent the failed bellows for valve 2CF-23, and a feedback bellows assembly that had been replaced but had not failed to Moore Productions for analysis as addressed by planned corrective action number 1. Moore Products reported no material or workmanship problems.

As a corrective action, General Office Nuclear Maintenance personnel are evaluating a digital positioner from Kytronics. They are testing the positioner under appropriate temperature conditions to consider the feasibility of replacing the Moore Products positioner with the Kytronics digital positioner. The resolution of this corrective action is being tracked by McGuire Problem Investigation Report 0-M89-0079.