Duke Power Company
McGuire Nuclear Generation Department
12700 Hagers Ferry Road (MG01VP)
Huntersville, NC 28078-8985

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

DUKE POWER

DATE: October 27, 1995

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

Subject: McGuire Nuclear Station Unit 1
Doc et No. 50-369
Licensee Event Report 369/95-05, Revision 0
Problem Investigation Process No.: 1-M95-1767

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 369/95-05, Revision 0, concerning a Unit 1 Manual 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,

TCMcMeekin

T.C. McMeekin

RJD/bcb

Attachment

cc: 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. Victor Nerses
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. George Maxwell
NRC Resident Inspector
McGuire Nuclear Station

310042

IE22

bxc: B.L. Walsh (EC11C)
Zack Taylor (CNS)
G.A. Copp (EC050)
J.I. Glenn (MG02ME)
P.R. Herran (MG01VP)
Cindy Davis (MG01CP)
J. E. Burchfield (ONS Reg Compliance)
G.H. Savage (EC06E)
G.B. Swindlehurst (EC11-0842)
M.S. Tuckman (EC07H)
R.F. Cole (EC05N)
R. M. Frye (EC05N)
Tim Becker (PB02L)
P.M. Abraham (EC08I)
R. B. White (MG01VP)
L.V. Wilkie (ON03SR)
D.P. Kimball (CN05SR)
NSRB Support Staff (EC05N)
Kay Crane (MG01RC)
Rich Casler (EC05N)

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMD NO. 3150-0104
EXPIRES: 5/31/95
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station Unit 1	05000 369	1 OF 7

TITLE (4)
A Unit 1 Manual Reactor Trip Was Initiated As A Result Of An Equipment Failure Caused By 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 NAME	DOCKET NUMBER(S)
										05000
09	27	95	95	- 005	- 00	10	27	95	N/A	05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (Check one or more of the following) (11)							
POWER LEVEL (10)	100%		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	
R. J. Deese, Manager, Safety Review Group	(704)	875-4065

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	HBC	VALVOP	C311	Yes					

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

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

Unit Status: Unit 1 - Mode 1 (Power Operations) at 100 percent power.

Event Description: On September 27, 1995, at 0923, a manual Reactor Trip was initiated on Unit 1 after valve 1SM-0007AB, Main Steam Isolation Valve Steam Line A, moved to the fail safe (closed) position. Operations (OPS) personnel quickly realized that valve 1SM-0007AB was closed, and responded in a proactive and conservative manner to manually trip the Unit prior to receiving an Automatic Reactor Trip. All systems responded as required after the trip. Unit 1 was returned to service on September 30, 1995, at 0410.

Event Cause: A cause of Unknown, possible equipment failure, has been assigned. Extensive investigation efforts have been unable to determine an exact mode of failure for the equipment.

Corrective Action: As a conservative measure, the associated fuses, solenoid valve coils, some wiring terminations, and relays were replaced to eliminate recurrence due to any possible intermittent problem with any of these components which could have been undetected during the troubleshooting efforts.

EVALUATION:

Description of Event

On September 27, 1995, Unit 1 was in Mode 1 (Power Operation) at 100 percent power. Train B of the Solid State Protection System (SSPS) [EIIS:JC] testing was in progress and Chemical And Volume Control (NV) system [EIIS:CB] Centrifugal Charging Pump [EIIS:P] 1B, 1NVPU0016 was removed from service for maintenance.

At 0923, Operations (OPS) Control Room [EIIS:NA] personnel received annunciator [EIIS:ANN] alarms [EIIS:ALM] for Steam Generator (SG) [EIIS:SG] 1A Flow Mismatch Low Steam Flow and 1A SG Main Steam Isolation Valve (MSIV) [EIIS:ISV] closed.

- The Reactor Operator at the controls (ROATC) immediately noted that the Main Control Board [EIIS:MCBD] indication for MSIV 1SM-0007AB, had moved to the fail safe (closed) position.

- The ROATC announced this to the other OPS Control Room personnel and then attempted unsuccessfully to manually reopen the valve by depressing the OPEN push-button on the Main Control Board.

- The Senior Reactor Operator (SRO) then instructed the ROATC to manually trip the Reactor [EIIS:RCT] prior to receiving an automatic Reactor Trip.

- The ROATC manually tripped the Reactor at 0923:42. An automatic Turbine Generator [EIIS:TG] Trip followed the Reactor trip.

- OPS personnel entered procedure EP/1/A/5000/E0, Reactor Trip Or Safety Injection, and then entered procedure EP/1/A/5000/ES-0.1, Reactor Trip Response.

- OPS personnel subsequently entered procedure OP/1/A/6100/05, Unit Fast Recovery. At this time the SSPS equipment was returned to normal alignment from the test configuration.

- The 4 hour notification to the NRC was made at 1023 in accordance with procedure RP/0/A/5700/10, NRC Immediate Notification Requirements.

- OPS, Maintenance, and Engineering personnel began an investigation to determine the cause of the event. After initial investigation and electrical troubleshooting of the circuits and components associated with MSIV 1SM-0007AB was completed, no apparent cause was found.

Subsequently, a more formal failure analysis was begun. Personnel from Engineering, Maintenance, Operations, and Safety Assurance met and a detailed listing of possible failure modes, both electrical and mechanical, was developed. Discussion was then held to determine failure modes that could be eliminated by information already known about the valve closure, and methods to test the other failure modes that had been identified to determine if they were true failure causes.

- The main testing activity determined at this meeting was to attempt reopening the valve manually by operating the OPEN push-button on the Main Control Board

- Personnel were placed to observe, and measure where possible, key parameters during the reopening attempt.

- The attempt was then made and the valve was successfully reopened from the OPEN push-button on the Main Control Board. Operation of the valve was completely normal from all observations and readings during this opening.

The failure investigation team was reconvened after this evolution to determine the failure modes eliminated or proved by this test. The Plant Operations Review Committee (PORC) was present during this review meeting.

- The results of this analysis determined that all identified possible failure modes had been tested and were eliminated with the exception of intermittent failure mode possibilities that did not recur during the test. Also, the possibility existed of some unrecognized interaction with the SSPS testing that had been in progress during the closure event.

- It was decided that several actions needed to be taken in order to address these possibilities. These were:

    1. Recreation of the SSPS testing in progress at the time of the original event and then cycling the valve to determine any unrecognized effect from the SSPS testing.

2. Thermography inspection of the associated terminal cabinets and junction boxes to identify any loose connections (hotter than normal connections) as potential causes of intermittent opening, coupled with additional tightness checking of terminations that could have caused this valve to close if they were broken intermittently.

3. An additional review by Electrical Engineering personnel to confirm that SSPS testing in progress at that time could not have caused the valve to close.

4. Conservatively replacing the fuses [EIIS:FU], some wiring terminations, relays [EIIS:RLY], and coils [EIIS:CL] on the solenoid valves [EIIS:FSV] associated with this valve, and testing of the removed components. (Replacement of these components, that could possibly be causes by intermittent problems, would allow more thorough testing of components that had been associated with this type event in the past.)

The PORC then met to evaluate restart of the Unit. A decision was made to allow restart with the approval of the Station Manager, contingent upon; (a) completion of the additional activities as discussed during the previous meeting and (b) provided those activities identified no further possible root causes of the event or previously unidentified Nuclear Safety concerns. It was further stipulated that if any concerns were so identified by those activities, the PORC would be reconvened to re-evaluate the restart decision.

- No root cause was found as a result of the additional testing, and based on the decision of the PORC, the Unit was returned to Mode 1, on September 30, 1995, at 0410, with the approval of the Station Manager.

Conclusion

This event did not result in any uncontrolled releases of radioactive material, personnel injuries, or radiation overexposures. The event is Nuclear Plant Reliability Data System (NPRDS) reportable due to the possible failure of the valve operator for valve 1SM-0007AB.

- The inadvertent movement of MSIV 1SM-0007AB to the fail safe (closed) position, has been assigned a cause of Unknown, possible equipment failure, since extensive investigation efforts have been unable to determine an exact mode of failure for any of the equipment.

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION(6-89)

APPROVED OMB NO. 3150-0104
EXPIRES 5/31/95

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
McGuire Nuclear Station, Unit 1	05000 369	95	05	0	5 OF 7

- All identified possible failure modes have been tested and were eliminated as valid causes with the exception of possible intermittent failure modes that did not recur during the follow up testing

- No problems which might have caused the inadvertent closure were found with the fuses, coils, solenoid valves, wiring, circuitry hardware, or logic associated with this valve. However, one of the relays removed did not perform in a manner that was expected for normal operation. Therefore, a possible cause for the event is considered to be failure of the SMAR7 relay contacts 1-1a, associated with the seal in circuitry of the Main Control board push-button for the valve.

- Testing has revealed that these contacts could be taken to the open circuit condition, which would have resulted in the closing of valve 1SM-0007AB. However, it is also suggested by the testing, that external influence on this relay would have to occur to initiate this condition. Such external influence could have been accomplished by bumping or jarring of the terminal cabinet in which the relay is located, or of the relay itself. No evidence has been found that this had happened at the time of the valve closure. No activities found to be in progress at that time would have caused bumping or jarring of the relay in question. It was also noted that even though the opening of these contacts would have caused the valve to move to the closed position, it would not have prevented the valve from reopening when the ROATC first attempted to do so from the OPEN push-button on the Main Control Board. These facts would refute the failure of the relay contacts as the cause of the valve closure.

A review of the Operating Experience Program (OEP) and Problem Investigation Process (PIP) data bases for the past 24 months revealed no similar reportable events associated with inadvertent movement of MSIVs to the fail safe (closed) position due to an unknown cause. This event is not considered to be recurring.

CORRECTIVE ACTION:

Immediate:

1. OPS personnel attempted to reopen MSIV 1SM-0007AB by operating the OPEN push-button on the Main Control Board.

2. OPS personnel initiated a Manual Reactor Trip.

3. OPS personnel entered procedure EP/1/A/5000/E0, Reactor Trip Or Safety Injection, and then entered procedure EP/1/A/5000/ES-0.1, Reactor Trip Response.

Subsequent:

1. OPS personnel entered procedure OP/1/A/6100/05, Unit Fast Recovery.

2. OPS and Maintenance personnel returned the SSPS B Train equipment to normal alignment from the test configuration.

3. Personnel from Engineering, Maintenance, OPS, and Safety Assurance performed a detailed investigation of possible failure modes, both electrical and mechanical.

4. OPS personnel successfully reopened the valve from the Main Control Board.

5. The fuses, relays, and coils from the solenoid valves associated with this valve were conservatively replaced and sent to the Duke Power Company Metallurgy Laboratory for further testing.

6. Electrical Engineering personnel performed additional reviews to confirm that SSPS testing then in progress could not have caused to valve to close.

Planned:

1. The Main Control Board push-button switch for valve 1SM-0007AB will be replaced during outage 1EOC10, and submitted to the Duke Power Company Metallurgy Laboratory for further testing.

SAFETY ANALYSIS:

Based on this analysis, this event is not considered to be significant. At no time were the health and safety of the public or plant personnel affected by this event.

- The Unit 1 Reactor was manually tripped due to the flow mismatch to SG 1A prior to reaching any automatic trip setpoint.

- A Turbine Generator Trip was automatic as a result of the manual Reactor Trip.

This type of trip is bounded by events as described in Chapters 15.2.4, "Inadvertent Closure Of Main Steam Isolation Valves", and 15.2.7, "Loss Of Normal Feedwater Flow", of the McGuire Final Safety Analysis Report (FSAR). The event described in Chapter 15.2.4, is more limiting because it assumes a complete loss of Main Feedwater. The CA system is assumed to provide decay heat removal capability following an automatic Reactor Trip from such a loss of flow.

- The MSIV involved in this event failed to the fail safe (closed) position.

- The Main Feedwater (CF) system [EIIS:SJ] was available after the trip and continued to provide feedwater flow.

- The Auxiliary Feedwater (CA) system [EIIS:BA] Train A Motor [EIIS:MO] Driven pump and the Turbine Driven Auxiliary Feedwater pump started on low-low level in SG 1A, and operated properly to assist in returning SG water levels to normal. The Train B Motor Driven pump did not start due to the Train B alignment for the then ongoing SSPS testing.

- Main Steam Line Code Safety Valves [EIIS:RV] 1SV-0020, 21 and 22 lifted and operated properly to relieve pressure for Loop A Steam Line following the trip. Loop A Steam Line pressure reached 1194 psig. Although valve 1SV-0022 lifted prior to the setpoint of 1205 psig, it was well within allowable tolerance. SG Power Operated Relief Valve, 1SV-0019, also opened and closed properly. All Steam Dump To Condenser valves operated properly. No Atmospheric Dump Valves opened. No Pressurizer [EIIS:PZR] Code Safety or Pressurizer Power Operated Relief Valves opened, nor were the setpoints for these valves reached.

- The Primary and Secondary plant parameters were stabilized at no load conditions within 30 minutes following the trip and all plant equipment responded as expected.