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 0 4

TITLE (4) Failure On A Printed Circuit Controller Card Cause Steam Generator Feed Regulator Valve To Close Resulting In Turbine And Reactor 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 4	1 6	8 8	8 8	— 0 0 7 —	0 0	0 5	1 6	8 8	N/A	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 (AREA CODE)
Steven E. LeRoy, Licensing	7 0 4 3 7 3 - 6 2 3 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 B	V A L V	C 6 0 0	Y					
X	S J	V A L V	P 0 3 2	N					

SUPPLEMENTAL REPORT EXPECTED (14)

X YES (If yes, complete EXPECTED SUBMISSION DATE) ☐ NO

EXPECTED SUBMISSION DATE (15): MONTH 0 7 DAY 0 1 YEAR 8 8

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

On 04/16/88 at 1033, valve 1CF-20, Steam Generator 1C Feedwater Regulating Valve, closed because of an unknown fault on the associated controller card in a control cabinet. Operations (OPS) recognized that Unit 1 would trip automatically so the Turbine Generator and Reactor were manually tripped. All systems and equipment responded as expected following the trip except that valve 1SV-7, Steam Generator 1C Power Operated Relief, opened approximately 10 psi lower than its setpoint band, and the open limit switch for valve 1SA-49, Steam Supply to Auxiliary Feedwater Pump Turbine, failed to actuate. OPS implemented the Reactor Trip and the Unit Fast Recovery procedure. Instrumentation and Electrical (IAE) replaced the faulty controller card, and Unit 1 was returned to service on 04/17/88 at 0501. This event is assigned a cause of Other since the component failure on the control card for valve 1CF-20 was due to unknown cause. IAE will study the failure of the open limit switch on valve 1SA-49 and determine the implications to other environmentally qualified equipment. The failed controller card will be analyzed by Westinghouse and a revision to this report will be submitted describing the test results.

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INTRODUCTION:

On April 16, 1988 at 1033, valve 1CF-20, Steam Generator 1C Feedwater Regulating Valve [EIIS:FCV], closed because of an unknown fault on the associated controller card in a control cabinet. Operations (OPS) personnel recognized that Unit 1 would soon trip automatically so the Turbine Generator and Reactor were manually tripped. All systems and equipment responded as expected following the trip except that valve 1SV-7, Steam Generator 1C Power Operated Relief [EIIS:RV], opened approximately 10 psi lower than its setpoint band, and the open limit switch for valve 1SA-49, Steam Supply to Auxiliary Feedwater Pump Turbine Valve [EIIS:FCV], failed to actuate. OPS personnel implemented the Reactor Trip procedure and the Unit Fast Recovery procedure. Instrumentation and Electrical personnel replaced the faulty controller card, and Unit 1 was returned to service on April 17, at 0501.

Unit 1 was in Mode 1, Power Operation, at 100% power when this event began.

This event has been assigned a Cause of Other because the component failure on the control card for valve 1CF-20 was due to an unknown cause.

EVALUATION:

Background

The Main Feedwater (CF) system [EIIS:SJ] includes a Feedwater Regulating Valve and an isolation valve in the flow path to each Steam Generator [EIIS:SG]. These valves are designed to fail in 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.

Description of Event

On April 16, 1988 at 1033, the power supply for valve 1CF-20 failed, causing the valve to close. This valve closure caused the level in Steam Generator 1C to decrease at about 25% per minute on the narrow range level instrumentation. OPS personnel received "Power Control Cabinet Failure" and "C Steam Generator Low Flow" alarms [EIIS:ALM] in the Control Room. Operations noticed that none of the indicator lights on the Unit 1 Main Control Board [EIIS:CBD] for valve 1CF-20 were lit, which indicated that control power for the valve had failed. OPS personnel realized that the unit could not be prevented from tripping on a Low Low Steam Generator Level signal, so OPS manually tripped the Turbine Generator [EIIS:TG] which tripped the Reactor and manually exercised the Reactor Trip Breakers [EIIS:MJB]. The CF Isolation Valves [EIIS:ISV] closed and the Auxiliary Feedwater (CA) system [EIIS:BA] started automatically. OPS personnel implemented procedure AP/1/A/5500/01, Reactor Trip. All other systems responded as expected and all parameters were approximately at their no-load values thirty minutes after the trip.

At 1131, OPS personnel made the required NRC notification according to RP/0/A/5700/10, NRC Immediate Notification Requirements procedure. At 1337, Performance (PRF) personnel notified OPS that the Alarm Summary indicated that valve 1SA-49 had not opened as required to supply steam to the CA Turbine Driven Pump [EIIS:P]. OPS personnel declared the CA Turbine Driven Pump inoperable, made an equipment failure update notification to the NRC, and initiated a work request to repair the valve.

IAE personnel assigned to determine why valve 1SA-49 did not operate, discovered that the open limit switch on the valve was not functioning because valve packing leakage had damaged the switch, but that the valve operated properly. IAE replaced the limit switch and completed functional tests by 2330. The limit switch was a Namco model EA180-34302. OPS personnel implemented procedure OP/1/A/6100/05, Unit Fast Recovery. Unit 1 was returned to service at 0501 on April 17, 1988.

Conclusion

IAE personnel investigating the loss of control power for valve 1CF-20 found the fuse [EIIS:FU] blown. IAE replaced the fuse and mounted the card in a test rack. The fuse immediately blew again, confirming that the component failure was on the Controller Card. Past preventative maintenance could not have predicted this failure because of the binary nature of such failures. IAE personnel replaced the Controller Card and sent the defective card to Westinghouse for failure mode analysis. The Controller Card was a Westinghouse supplied Model No. 2837A16G03.

The manual Turbine Generator and Reactor Trip were initiated because an unknown failure on the 7300 System Controller Driver card occurred which caused the power supply fuse to open and deenergize control power to valve 1CF-20. Therefore, this event has been assigned a Cause of Other.

One other anomaly occurred during this event. Valve 1SV-7 opened approximately 10 psi below its setpoint band. PRF personnel initiated a work request to investigate and/or repair the valve.

A review of McGuire Licensee Event Reports (LERs) revealed 21 other Reactor Trips caused by equipment malfunctions. Three of these equipment malfunctions led to low feedwater flow transients. LER 50-370/85-18 concerned an event during which a solenoid valve failure caused a Feedwater Isolation Valve to close and a subsequent Reactor Trip. LER 50-369/86-07 concerned a Reactor Trip caused by a mechanical failure of a fuse on a power supply card for a Feedwater Isolation Valve. LER 50-369/86-02 concerned a Reactor Trip on a Low-Low Steam Generator Level signal caused by a failure of the automatic portion of the Feed Regulating Controller Card. Corrective actions for these three events were specific for the particular event. Since there have been previous similar events with the same root cause, this event is considered recurring; however, corrective actions for these previous events probably could not have prevented this event.

The only failure which is reportable to the Nuclear Plant Reliability Data System (NPRDS) is the failure of valve 1SV-7. Valve 1SV-7 is a Control Components Model POR-002.

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

CORRECTIVE ACTIONS:

Immediate: OPS personnel implemented procedure AP/1/A/5500/01, Reactor Trip.

Subsequent:

1) IAE personnel replaced the faulty Control Card supplying power to valve 1CF-20.

2) IAE personnel replaced the faulty open limit switch on valve 1SA-49.

3) OPS personnel completed procedure OP/1/A/6100/05, Unit Fast Recovery.

4) IAE personnel sent the faulty Controller Card to Westinghouse for failure mode analysis.

Planned:

1) An addendum to this report to describe the results of the Westinghouse investigation of the cause and location of the Controller Card failure will be submitted to NRC.

2) IAE will study the failure of the open limit switch on valve 1SA-49 and determine the implications to other environmentally qualified equipment.

SAFETY ANALYSIS:

This event is bounded by the McGuire Final Safety Analysis Report (FSAR) Section 15.2.7., Loss of Feedwater Flow Transient. This analysis predicted that the CA system would be capable of removing sufficient energy from the Reactor Coolant system [EIIS:AB] so that no Reactor Coolant would be lost from Safety Valves. In this event, neither the Pressurizer Power Operated Relief Valves [EIIS:RV] nor Code Safety Valves [EIIS:V] were required or actuated. All systems performed as expected following the Reactor Trip.

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

This event is considered to be of no significance with respect to the health and safety of the public.

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

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

May 16, 1988

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

Subject: McGuire Nuclear Station, Unit 1
Docket No. 50-369
Licensee Event Report 369/88-07

Gentlemen:

Pursuant to 10CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 369/88-07 concerning a reactor trip that occurred on April 16, 1988. This report is being submitted in accordance with 10CFR 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,

H.B. Tucker JWH

Hal B. Tucker

SEL/20/sbn

Attachment

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