

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)

McGuire Nuclear Station, Unit 1

DOCKET NUMBER (2)

05000369

PAGE (3)

1 of 5

TITLE (4)

A Unit 1 Manual Reactor Trip Was Initiated As A Result Of An Equipment Failure

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIA ¹ NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
01	29	95	95	001	1	06	07	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
	20.405(a)(1)(iii)	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)	Abstract below and
	20.405(a)(1)(iv)	50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)	in Text, NRC Form
	20.405(a)(1)(v)	50.73(a)(2)(iii)		50.73(a)(2)(x)	366A)

NAME

R. J. Deese, Manager, Safety Review Group

TELEPHONE NUMBER

AREA CODE

(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
				NO					

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 29, 1995, at 1822, a manual Reactor Trip was initiated on Unit 1 after Steam Generator (S/G) Feedwater Isolation Valve 1CF-26 moved to its fail-safe (closed) position. Operations (OPS) personnel quickly realized valve 1CF-26 was going closed. OPS personnel responded in a proactive and conservative manner and manually tripped the Reactor prior to receiving an automatic Reactor Trip on low-low level in S/G 1D. All systems responded as required. Prior to the event, Unit 1 was in Mode 1 (Power Operation) at 100 percent power. Unit 1 was returned to service on January 31, 1995 at 0955. This event is assigned a cause of Equipment Failure. The valve closed due to loss of power caused by a failed fuse in the control circuitry to the solenoid. An analysis of the failed fuse by the Duke Power Company Metallurgy Lab indicated the fuse failure occurred by brittle fracture through the fusible alloy button adjacent to the copper wire lead to one fuse cap. Corrective actions to prevent recurrence include the replacement of this type of fuse in applications deemed critical to plant operation.

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NRC FORM 365A (89)		U.S. NUCLEAR REGULATORY COMMISSION(6-		APPROVED OMB NO. 3150-0104 EXPIRES:5/31/95	
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EVALUATION:

Description of Event

On January 29, 1995, at 1822, Unit 1 was operating in Mode 1 (Power Operation) at 100 percent power when Steam Generator (S/G) [EIS:SG] Feedwater Isolation Valve [EIS:ISV] 1CF-26 started going closed. Operations (OPS) personnel quickly realized valve 1CF-26 was going closed and tried unsuccessfully to open the valve by operating the OPEN push-button on the Main Control Board. OPS personnel manually tripped the Reactor [EIS:RCT] prior to receiving an automatic Reactor Trip on low-low level in S/G 1D.

OPS personnel entered procedure EP/1/A/5000/E-0, Reactor Trip or Safety Injection, and then entered procedure EP/1/A/5000/ES-0.1, Reactor Trip Response. The Auxiliary Feedwater (CA) system [EIS:BA] started on low-low level in S/G 1D and operated properly. OPS personnel subsequently entered procedure OP/1/A/6100/05, Unit Fast Recovery.

An investigation of the reasons for closure of valve 1CF-26 was conducted by Mechanical Maintenance (MM) and Instrument and Electrical (IAE) personnel. This investigation revealed that valve 1CF-26 closed due to a loss of power to the Train B safety solenoid [EIS:SOL]. Additionally, the control wiring for the Train A and Train B solenoids was found to be reversed.

The initial cause of valve 1CF-26 closure appeared to be a blown FNQ type fuse [EIS:FU] in the control circuitry for the Train B safety solenoid. Further investigation did not identify any electrical cause for the failed fuse. Subsequent evaluation revealed that the fuse had experienced a mechanical failure. This particular fuse had previously passed a resistivity screening criteria prior to initial installation in this application. The screening criteria, which was provided by the fuse manufacturer, was used for FNQ type fuses in critical applications to predict the fuse susceptibility to mechanical failure. The failure of this fuse represents the first and only known mechanical failure of a pre-screened FNQ type fuse at the station.

On January 31, 1995, at 0800, a Plant Operating Review Committee (PORC) meeting was held to discuss the Unit 1 restart. As a conservative measure, the PORC recommended the evaluation and replacement of any Unit 1 FNQ type fuses in applications deemed to be critical to plant operation.

Following replacement of the FNQ type fuses deemed critical to plant operation, Unit 1 was returned to service on January 31, 1995 at 0955.

Conclusion

This event is assigned a cause of Equipment Failure. The fuse that failed was a Bussman FNQ-3 type fuse located in the control circuitry of the Train B safety solenoid for valve 1CF-26. Since the control wiring for the Train A and Train B solenoids was reversed, failure of the fuse caused the Train A safety solenoid to lose power. Valve 1CF-26 moved to its safety position (closed) as designed.

The solenoid wiring configuration is an identical design allowing either of the solenoids to manipulate the shuttle valve in the hydraulics to close the isolation valve. Therefore, their safety function was not degraded. This condition was documented on Problem Investigation Process (PIP) 1-M95-0221.

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An FNQ type fuse consists of dual elements joined by a fusible alloy. One-half of the fuse is a "slow blow" (overcurrent) element while the other half is a "fast blow" (short circuit) element. The FNQ type fuse in question was disassembled by Engineering personnel. An examination of the fuse internals by Engineering personnel indicated the fuse element had separated at the fusible alloy which connects the "slow blow" and "fast blow" portions of the fuse element.

The failed fuse was sent to the Duke Power Company Metallurgy Lab for analysis which revealed the failure occurred by brittle fracture through the fusible alloy button adjacent to the copper wire lead to one fuse cap. The failure was not in the normal manner of melting of the fusible alloy button due to overload or high currents but through a mechanical failure of the porous layer of the fusible alloy next to the copper lead.

As a conservative measure, all FNQ type fuses in the control circuitry for Train A and Train B safety solenoids for valve 1CF-26 were removed and replaced with FLQ type fuses. IAE personnel properly connected the control wiring for the solenoids associated with this valve. All FNQ type fuses in the control circuitry for Train A and Train B safety solenoids for S/G Feedwater Isolation Valves 1CF-28, 1CF-30, and 1CF-35 were also removed and replaced with FLQ type fuses. Additionally, IAE personnel verified the control wiring to the solenoids for these valves was properly connected. All FNQ type fuses in the control circuitry for Train A and Train B safety solenoids for Main Steam (SM) [EII:SB] Isolation Valves 1SM-1, 1SM-3, 1SM-5, and 1SM-7 were removed and replaced with FLQ type fuses. OPS personnel, IAE personnel, and Engineering personnel determined that these were the applications utilizing FNQ type fuses that needed to be replaced prior to restart.

A review of the PIP data bases for the past 24 months revealed 5 events involving Reactor Trips in which the cause was an Equipment Failure. These events are documented in Licensee Event Reports (LERs) 370/93-01, 370/93-02, 369/93-05, 370/93-08, and 369/93-09. Two of these events, LER 370/93-01 and 370/93-02, involved Equipment Failures in the Main Feedwater (CF) system [EII:SJ]. However, the equipment failures in these incidences were caused by failures unrelated to equipment that failed during this event.

In addition to these LERs, one event occurred in 1992, LER 370/92-07, which was caused by the failure of a FNQ type fuse. Since this event involved a failure of an FNQ type fuse, this event is considered recurring. Corrective actions from that event included a testing process to determine which fuses might be susceptible to mechanical failure. The failed fuse that caused valve 1CF-26 to close had passed this test and was not considered to be a candidate for mechanical failure. This is the first mechanical failure of a FNQ type fuse that had passed this screening criteria. In addition, IAE personnel perform a monthly surveillance of all Nuclear Safety Related FNQ type fuses if their failure is not detectable by Control Room indication.

This event is not NPRDS reportable because the closure of valve 1CF-26 was due to failure of an associated device.

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

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CORRECTIVE ACTION:

Immediate:

1. OPS personnel tried to open valve 1CF-26 by operating the OPEN push-button.
2. OPS personnel initiated a manual Reactor Trip.
3. OPS personnel entered procedure EP/1/A/5000/E-0, Reactor Trip or Safety Injection.

Subsequent:

1. OPS personnel entered procedure EP/1/A/5000/ES-0.1, Reactor Trip Response.
2. OPS personnel entered procedure OP/1/A/6100/05, Unit Fast Recovery.
3. The evaluation of FNQ type fuses in Unit 1 critical applications has been completed.
4. FNQ type fuses have been replaced in Unit 1 applications that needed to be replaced prior to restart.
5. The evaluation of FNQ type fuses in Unit 2 critical applications has been completed.

Planned:

1. The remaining FNQ type fuses in Unit 1 critical applications will be replaced.
2. FNQ type fuses will be replaced in Unit 2 applications as determined by the evaluation.
3. Engineering personnel will evaluate the need for replacement of remaining FNQ type fuses, type of replacement, and current fuse surveillances.

SAFETY ANALYSIS:

The Unit 1 Reactor was manually tripped prior to reaching the S/G 1D Low-Low level setpoint. The Turbine Generator Trip was automatic as a result of the manual Reactor Trip. The Reactor Trip as a result of Low-Low S/G level is bounded by the "Loss of Normal Feedwater Flow" event of the McGuire Final Safety Analysis Report (FSAR), Chapter 15.2.7. The event described in the FSAR 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 Low-Low S/G water level.

The CF system was available after the Reactor Trip and continued to provide feedwater flow. The CA system started automatically as designed and provided additional feedwater flow, as necessary, to all 4 S/Gs to assist in returning S/G water levels to normal. Water level in S/G 1D reached a low point of 38.4 percent (wide range level indication) and began to recover almost immediately following the manual Reactor Trip. Water level in S/G 1D reached no load condition approximately 30 minutes after the manual Reactor Trip. The Reactor Coolant (NC) system [EII:AB]

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Power Operated Relief Valves (PORVs) [EIS:RV] and Code Safety Valves did not open and were not challenged.
The S/G PORVs and Code Safety Valves did not open and were not challenged.

This manual Reactor Trip presented no hazard to the integrity of the NC or Main Steam system. There were no
radiological consequences as a result of this event.

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