

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) McGuire Nuclear Station - Unit 1	DOCKET NUMBER (2) 0 5 0 0 0 3 6 9	PAGE (3) 1 OF 0 6
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TITLE (4)  
Reactor Trip on Low-Low Steam Generator Level Caused by a Failed Fuse

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 3	2 5	8 6	8 6	0 0 7	0 0	0 4	2 3	8 6			0 5 0 0 0
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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) 1 0 0	20.402(b)	20.405(a)(1)(i)	20.405(a)(1)(ii)	20.405(a)(1)(iii)	20.405(a)(1)(iv)	20.405(a)(1)(v)	20.406(c)	50.38(c)(1)	50.38(c)(2)	50.73(a)(2)(i)	50.73(a)(2)(ii)	50.73(a)(2)(iii)	50.73(a)(2)(iv)	50.73(a)(2)(v)	50.73(a)(2)(vi)	50.73(a)(2)(vii)(A)	50.73(a)(2)(vii)(B)	50.73(a)(2)(x)	73.71(b)	73.71(c)	OTHER (Specify in Abstract below and in Text, NRC Form 365A)
										<input checked="" type="checkbox"/>											

LICENSEE CONTACT FOR THIS LER (12)

NAME Julio G. Torre, Licensing	TELEPHONE NUMBER
	AREA CODE: 7 0 4   3 7 3 1 - 1 8 1 0 2 9

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
B	S/J	I F I V E I	1 2 1 0	NO					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)	<input checked="" type="checkbox"/> NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

On March 25, 1986, at 1748, the Unit 1 reactor tripped on a Steam Generator (S/G) 1B low-low level signal. Valve 1CF-30 - a hydraulic S/G Feedwater Containment Isolation Valve - closed at 1747 due to a failed indicating fuse in the valve control circuitry isolating the main feedwater to S/G 1B. The subject fuse was a 2 ampere indicating fuse produced by the Bussmann Division of McCraw Edison. Control room personnel attempted to reopen valve 1CF-30 without success. Control room personnel also started both motor-driven Auxiliary Feedwater pumps and simultaneously aligned feedwater flow through the auxiliary feed nozzle in an unsuccessful attempt to maintain S/G water level above the trip setpoint.

Unit 1 was in Mode 2 at 100% power when the reactor/turbine trips occurred. Plant systems responded as expected for the transient. No emergency core cooling systems were required during the incident.

All indicating fuses in the control circuitry of the Units 1 and 2 main feedwater isolation valves, and Unit 2 regulating and bypass valves will be replaced with non-indicating fuses during the current end of Cycle 2 refueling outage.

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

On March 25, 1986, at 1748, the Unit 1 reactor tripped on Steam Generator (S/G) 1B low-low level signal. Valve LCF-30 - a hydraulic S/G Feedwater Containment Isolation Valve - closed at 1747 due to a failed indicating fuse in the valve control circuitry isolating the main feedwater to S/G 1B. Control room personnel attempted to reopen valve LCF-30 without success. Control room personnel also started both motor-driven Auxiliary Feedwater pumps and simultaneously aligned feedwater flow through the auxiliary feed nozzle in an unsuccessful attempt to maintain S/G water level above the trip setpoint.

Unit 1 was in mode 1 at 100% power when the reactor/turbine trips occurred. Plant systems responded as expected for the transient. No emergency core cooling systems were required during the incident.

BACKGROUND:

The Main Feedwater (CF) [EIIS:SJ] system is designed to provide adequate feedwater flow to all four S/Gs at the temperature, pressure, and flow rate required to maintain proper S/G water level with respect to reactor power and turbine steam requirements. The CF system also provides feedwater isolation to prevent overpressurization of the containment building in the event of a main feedwater or main steam pipe rupture. Valve LCF-30 - a hydraulic S/G feedwater containment isolation valve - is designed to isolate the main feedwater to S/G 1B. This valve has two modes of operation: 1) Manual opening or closing and; 2) Automatic fast closing.

The Auxiliary Feedwater System (CA) [EIIS:BA] is provided as a backup for the CF system. It is designed as a means to dissipate heat from the Reactor Coolant System [EIIS:AB] when normal systems are not available. The CA system is provided with two 100 percent capacity motor driven pumps and one 200 percent capacity turbine driven pump. Each CA motor driven pump provides feedwater to two S/Gs. The turbine driven pump provides water to all four steam generators.

Main feedwater can also be introduced to the steam generator through the auxiliary feed nozzle - located in the shell of the S/G - by means of valving in a cross connection between main and auxiliary feedwater lines.

DESCRIPTION OF EVENT:

On March 25, 1986 at 1747, valve LCF-30 failed closed. Duke Power personnel attempted to reopen valve LCF-30 without success. They also manually started both motor-driven CA pumps and simultaneously aligned feedwater flow through the auxiliary feed nozzle in an unsuccessful attempt to maintain S/G water level above the trip setpoint. At 1748, the reactor tripped due to S/G 1B low-low level followed by a turbine trip. Reactor trip and immediate notification procedures were implemented as required.

The turbine driven auxiliary feedwater pump was started approximately 2 minutes after the reactor trip to help recover level in steam generator B. Level was above 34% in all four steam generators within 30 minutes after the reactor trip.

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Reactor Coolant temperature and pressure recovered to its expected no-load values within 30 minutes after the trip. No emergency core cooling systems were required during this event.

After the reactor trip, a work order was written to investigate the problem with valve LCF-30. At approximately 2200, it was determined that valve LCF-30 automatically closed due to a failed fuse in the control circuitry. The control circuitry and the motor operator of valve LCF-30 were inspected and no other problems were discovered.

CONCLUSION:

The failed fuse in the control circuitry of valve LCF-30 was a 2 ampere, indicating fuse (Part No. FNA-2) produced by the Bussmann Division of McGraw Edison. The failure of the fuse was due to the element having pulled loose from the solder joint inside the fuse. The solder joint was not broken but instead, the element wire pulled out of the joint. This type of failure may suggest a cold, insufficient solder joint.

During the current end of cycle 2 Unit 2 refueling outage, a systematic program will be implemented to replace all indicating fuses with non-indicating fuses. The CF isolation valves, regulating valves, and bypass valves, along with valves which have caused reactor trips in the past due to this type of fuse failure, will have top priority.

This incident was attributed to the failure of the Bussmann 2 ampere indicating fuse in the control unit of hydraulic valve LCF-30. A possible contributory cause is a Manufacturing Deficiency. In January, 1982, and May, 1983, a Bussmann representative speculated that the fuse failures were isolated cases and were not a generic problem. The Bussmann representative recommended that the fuses in lot numbers which showed a high incidence of failures be segregated. Due to the way installation of items such as Bussmann fuses are handled, it is difficult to determine which manufacturer lot number certain fuses came from. Therefore, it is unknown if all the failed fuses are from one manufacturer lot number. McGuire Nuclear Station personnel examined Bussmann indicating fuses in stock and found that an average of 8 percent of the fuses were indicating open (failed). The fuses examined were from several different manufacturer lot numbers and the failures within lot numbers ranged from 0 percent to 33 percent.

A review of past events indicated several incidents involving blown or failed Bussmann fuses. Two of these incidents (Licensee Event Reports 369-81-179 and 369-85-36) involved reactor trips attributed to failed Bussmann indicating fuses. Therefore, this is considered a recurring event.

Although the failed Bussmann fuse is not reportable under the Nuclear Plant Reliability Data System (NPRDS), a search was performed on component failures due to defective or loose parts. Nothing was found involving Bussmann fuses. A search of valve failures at McGuire was also performed. Of the nearly 300 valve failures listed, none involved blown or failed Bussmann indicating fuses.

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

There were no performance anomalies associated with this reactor trip.

There were no personnel injuries, radiation exposures, or releases of radioactive material resulting from this incident.

CORRECTIVE ACTIONS:

- Immediate:
- 1) Duke Power personnel attempted to reopen valve ICF-30.
  - 2) Both motor-driven CA pumps were manually started.
  - 3) Reactor trip and immediate notification procedures were implemented.
- Subsequent:
- 1) The failed fuse in the control circuitry of valve ICF-30 was replaced.
  - 2) All indicating fuses in the control circuitry of all Unit 1 CF isolation valves were replaced with Bussmann non-indicating fuses.
- Planned:
- A program to systematically replace all indicating fuses with non-indicating fuses throughout the plant is under development.
- All indicating fuses in the control circuitry of the Unit 2 CF isolation valves, regulating valve, and bypass valves will be replaced with non-indicating fuses during the current end of cycle 2 refueling outage.

SAFETY ANALYSIS:

The reactor tripped due to S/G 1B low-low level. No emergency core cooling systems were required during the incident. Adequate core cooling was maintained at all times. The reactor pressure boundary was not challenged. Valve ICF-30 failed to its safe position (closed). No other safety system's availability was affected by this incident.

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

TRANSIENT ASSESSMENT:

Prior to the reactor trip, reactor power decreased due to the increase in average coolant temperature that resulted from the isolation of S/G B. Following the reactor trip power dropped quickly, as expected, to its decay heat level.

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Pressurizer pressure peaked prior to the reactor trip at 2239 psig. This is a very minor increase above its reference value of 2235 psig. Following the reactor trip, pressure decreased to a minimum value of 2024 psig before recovering. Pressure had returned to its reference value within 30 minutes after the reactor trip.

Reactor coolant loop average temperature increased about 2 degrees F in loop B prior to the reactor trip due to the decrease in steam generator heat transfer and the resulting increase in cold leg temperature. Following the reactor trip, temperature dropped immediately to 562 degrees F as reactor power decreased. Temperature then slowly decreased to its post-trip minimum of about 553 degrees F thirteen minutes after the reactor trip. Auxiliary feedwater flow was reduced to the steam generators at that time. Temperature then recovered to its expected no-load value of 557 degrees F within 30 minutes after the reactor trip.

Pressurizer level changed little pre-trip. Following the trip, level dropped immediately to 38% and then decreased as Tave did. Level reached its minimum value of 22.3% about 10 minutes after the reactor trip. At that time the charging flow was increased and level began to recover. Level had stabilized at its no-load target by thirty minutes after the trip. Charging flow was decreased as level approached the target. Letdown flow remained almost constant throughout the event. There was no emergency boration flow during this event.

Reactor coolant loop flow increased slightly due to the coolant density changes. No reactor coolant pumps were tripped during the event.

Steam pressure increased following the turbine trip as expected. Peak pressure was 1104 psig in steam generator C. This is below the steam generator PORV setpoint of 1125 psig, and none of the PORVs opened. Steam pressure then decreased as auxiliary feedwater was being fed into the steam generators. Pressure leveled off about 10 minutes after the trip, as auxiliary feedwater flow to steam generator B was reduced. Pressure was about 1040 psig at that time. Pressure then began to increase once flow to the other three steam generators was decreased. Pressure had recovered to 1076 psig within 30 minutes after the trip. This is slightly below the no-load target of 1092 psig, but within normal limits. Main steam flow responded as expected for a trip from this power level.

Steam generator B narrow range level dropped rapidly following closure of the main feedwater isolation valve. Level dropped to the trip setpoint in less than a minute. Level increased slightly in the other 3 generators as the flow from B was diverted to them. Level went off scale low in steam generator B following the reactor trip. Level dropped to about 30% in the other three generators. Level was recovered using auxiliary feedwater. Level was above 34% in all four steam generators within 30 minutes after the reactor trip.

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Steam generator feedwater flow dropped to zero in steam generator B as the isolation valve closed. Flow increased to the other generators as the flow redistributed itself. The slight increase in flow pre-trip occurred as the operators opened the feedwater regulating valve bypass valve and aligned flow to the upper feedwater nozzle. Flow was decreasing to the other three generators as the level control system closed the regulating valves. Flow dropped to zero in all four steam generators following the trip as feedwater was isolated on reactor trip with low Tave.

The motor driven auxiliary feedwater pumps were started ten seconds prior to the reactor trip in an effort to maintain level in steam generator B. Flow was reduced below 250 gpm per generator shortly after the reactor trip. The turbine driven auxiliary feedwater pump was started approximately 2 minutes after the reactor trip to help recover level in steam generator B. A higher flow rate to this generator was maintained. The turbine driven pump was secured about eight minutes later, as wide range level was recovering. About three minutes later flow was again reduced to the A, C, and D steam generators. Flow was maintained at a higher level in steam generator B until it had returned to its no-load target. Flow to all four generators was then reduced. Auxiliary feedwater flow was properly controlled by the operators and there was little overcooling.

Plant personnel responded appropriately to the event. Due to the nature of the failure, a reactor trip could not be avoided.

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HAL B. TUCKER  
VICE PRESIDENT  
NUCLEAR PRODUCTION

April 23, 1986

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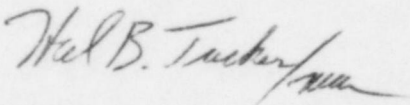
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Subject: McGuire Nuclear Station, Unit 1  
Docket No. 50-369  
LER 369-86-07

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a)(2)(i)(IV), attached is Licensee Event Report 369-86-07 concerning a McGuire Unit 1 Reactor Trip on Low-Low Steam Generator Level Caused by a Failed Fuse. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,



Hal B. Tucker

JGT/jgm

Attachment

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