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 2	0 5 0 0 0 3 7 0	1 OF 0 3

TITLE (4) Reactor Trip due to reverse flow through tempering flow check valve causing steam Generator low-low level.

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 5	1 1	8 4	8 4	0 1 3	0 0	0 6	1 4	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 2 3

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)	
Phillip B. Nardoci, Licensing Engineer	7 0 4	3 7 3 - 7 4 3 2

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 C	V	K 0 8 5	Y					

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 May 11, 1984, while returning to power following a trip (Ref. LER 370/84-12), a reactor trip occurred at 0051 when steam generator (S/G) 2D level dropped below the S/G low-low level reactor trip setpoint. The S/G low-low level condition resulted from feedwater draining to the main condenser as reverse flow purge on S/G 2D was attempted. The flow path of water was through check valve 2CF-158, D S/G Tempering Flow Inlet Check, whose failure resulted in flow in the reverse direction. The unit was then stabilized and restarted with valve 2CF-158 isolated. Unit 2 was in Mode 1 at 23% power at the time of this event. This incident is attributed to Component failure because valve 2CF-158 failed open allowing reverse flow.

Valve 2CF-158 will be repaired during the next extended outage.

IE22
1/1

NRC Form 366
(9-83)

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 4	0 1 3	0 0	0 2 OF 0 3

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

On May 11, 1984, while returning to power following a trip (Ref. LER 370/84-12), a reactor trip occurred at 0051 when steam generator (S/G) [EIIS:GEN] 2D level dropped below the S/G low-low level reactor trip setpoint. The S/G low-low level condition resulted from feedwater [EIIS:SJ] draining to the main condenser [EIIS:SG] as reverse flow purge on S/G 2D was attempted. The flow path of water was through check valve [EIIS:V] 2CF-158, D S/G Tempering Flow Inlet Check, whose failure resulted in flow in the reverse direction. The unit was then stabilized and restarted with valve 2CF-158 isolated. Unit 2 was in Mode 1 at 23% power at the time of this event. This incident is attributed to Component Failure because valve 2CF-158 failed open allowing reverse flow.

The steam generators (S/G) are fed from feedwater (CF) through the upper (auxiliary) nozzles [EIIS:PSP] until the final feedwater temperature is 250°F and CF flow to each S/G is greater than 17% full flow. At this point of operation the S/G lower (main) nozzles are preheated (reverse flow purge) to allow transfer from the upper nozzle.

At 0035 on May 11, 1984, valve 2CF-137A (D S/G CF Temper Isolation) was opened; and valve 2CF-157B (D S/G CF Temper Isolation) was opened at 0036. Valve 2CF-187, Tempering Back-flow from CF Nozzles, was then opened and Operations personnel subsequently noticed S/G 2D level decreasing and 2D feedwater regulator valve (2CF-17AB) full open. Operations personnel took manual control of 2CF-17AB and positioned it to a more restrictive position (closed slightly to decrease the level shrinkage in S/G 2D), in anticipation of S/G level decrease due to CF shrinkage. (Shrinkage and swell is the change in level due to a change in average density of the water/steam mixture in the S/G). The level continued to decrease so operations personnel adjusted 2CF-17AB (opened manually to increase feedwater flow) so that feedwater flow was greater than steam flow. At 0049, the main steam [EIIS:SB] bypass valves, 2SB-3, 2SB-9, 2SB-12, 2SB-21, and 2SB-24 were opened slightly to initiate swelling of S/G 2D level. However, S/G 2D low-low level reactor trip setpoint [EIIS:JC] was reached at 0051 and the reactor and turbine/generator both tripped.

An investigation of the reactor trip determined that a check valve (2CF-158, D S/G Tempering Flow Inlet Check) in the tempering (heatup) line [EIIS:SC], which is connected to the reverse purge piping, stuck open allowing a 2 inch flow path from S/G 2D to the main condenser with no orifice restrictions. The failure of 2CF-158 is consistent with the S/G 2D level decrease, feedwater flow increase, and condenser level increase. Operator response to the event was rapid and exceptionally perceptive in maintaining the 2D level above the trip setpoint for eleven minutes with the excessive leak. This feedwater flow path split and approximately 5% of the flow went to the condenser. The operators responded to the feedwater transient as a feedwater regulator [EIIS:JB] problem, not a feedwater leak. (The main feedwater regulator valve, 2CF-17, had had problems the previous day (Ref. LER 370/84-12)). 2CF-158 is a 2 inch check valve manufactured by Kerotest, which has a spring loaded piston mounted at an angle to the flow. It is theorized that the plunger of this valve either was cocked or that material blocked it so that reverse flow was possible.

Valve 2CF-158 was isolated by closing valve 2CF-186, S/G D Temper Flow Isolation. The unit was successfully restarted with 2CF-158 isolated. 2CF will be repaired during the next extended outage.

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 4	— 0 1 3	— 0 0	0 3 OF 0 3

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

A restrictive procedure change was written to close 2CF-186 prior to reverse purge, and re-open 2CF-186 for tempering flow to the upper nozzles.

When the main steam bypass valves [EIIS:V] were opened prior to the reactor trip, primary pressure and temperature decreased 40 psig and 4°F, respectively. Pressure and temperature turned around as reactor power increased because of the combined actions of the moderator temperature coefficient and the reactor control system. At the time of the trip, pressurizer pressure and temperature were 2237 psig (reference pressure 2235 psig) and 561°F (reference temperature 563°F).

The reactor tripped on anticipated loss of one out of four primary heat sinks (S/Gs). Operators isolated the leak shortly after the trip, which ensured that all four S/Gs were available for cooling primary water. Had 2D S/G been disabled (boiled dry) by an excessive leak, the other three S/Gs would still have been available for reactor cooling. Reactivity was properly controlled by the reactor trip. Pressurizer pressure decreased to a minimum of ~2180 psig before recovering. Pressure overshot slightly to 2280 psig before settling out near its reference value of 2235 psig. The pressurizer Power operated relief valve and code safety valves were not challenged. Primary average temperature decreased following the trip, reaching a minimum of ~545°F before recovering. This was about 10°F lower than normal as a result of lower than normal post-trip steam pressure. Pressurizer level decreased following the trip to a minimum value of ~22%. Level recovered above its no load target of 25% within 15 minutes after the trip. Letdown was not isolated.

Steam pressure was ~50 psig lower than normal at the time of the trip because the steam dumps [EIIS:SB] had been opened to swell steam generator levels. Steam pressure peaked at 1040 psig after the trip, which is below its expected post-trip value of 1092 psig. Steam pressure decreased to a minimum value of ~970 psig following the trip before recovering. Steam pressure was lower than normal after the trip because of the lower power level prior to the trip and the lower than usual pre-trip steam pressure. Pressure remained well above the main steam isolation valve closure setpoint. The steam generator power operated relief valves and main steam safety valves were not challenged. The narrow range level in steam generators A, B, and C decreased to a minimum of ~25% before being recovered with auxiliary feedwater. Level was recovered to near its expected no load value of 38% within thirty minutes after the trip. The narrow range level in steam generator D went offscale low when the steam voids collapsed following the trip and remained offscale for ~15 minutes after the trip. Level had been decreasing prior to the trip because of the feedwater diversion. Level was recovered when sufficient mass was added by the auxiliary feedwater system [EIIS:BA], and was above 25% within thirty minutes after the reactor trip.

Main feedwater was isolated as expected following the trip on reactor trip on coincident low Tave. Auxiliary feedwater was initiated on steam generator low-low level and was used to recover level.

No Engineered Safety Features were demanded. No abnormal reactor coolant leakage or radioactivity release occurred as a result of this transient. The primary cooldown rate was within the Technical Specification Limit. The health and safety of the public were unaffected by this incident.

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

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

June 14, 1984

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

Subject: McGuire Nuclear Station, Unit 2
Docket No. 50/370
LER 370/84-13

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 370/84-13 concerning a Unit 2 reactor trip due to reverse flow through a tempering flow check valve causing a steam generator low-low level which is submitted in accordance with §50.73(a)(2)(iv). Initial notification of this event was made (pursuant to §50.72 Section (b)(2)(ii)) with the NRC Operations Center via the ENS on May 11, 1984. This event was considered to be of no significance with respect to the health and safety of the public.

In order to provide complete information relating to the event this report is being submitted three working days late. We regret any inconvenience this may have caused.

Very truly yours,

H.B. Tucker / PBN

Hal B. Tucker

PBN:glb
Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Suite 2900
101 Marietta Street, NW
Atlanta, GA 30323

Records Center
Institute of Nuclear Power Operations
1100 Circle 75 Parkway, Suite 1500
Atlanta, Georgia 30339

Mr. W. T. Orders
NRC Resident Inspector
McGuire Nuclear Station

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

IE22
1/1