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 5

TITLE (4)

Manual Reactor Trip Following Feedwater Pump 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)
~~1 2~~ 1 1	~~1 7~~ 1 5	8 4	8 4	0 2 9	0 0	1 2	1 7	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	0 2 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)
Scott Gewehr - Licensing	7 0 4 3 7 3 - 7 5 8 1

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

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[ ] YES (If yes, complete EXPECTED SUBMISSION DATE)	[XX] NO				

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

On November 15, 1984, a Manual Reactor Trip was initiated on Unit 2 following the loss of both feedwater pumps, with one feedwater pump isolated for maintenance, the second pump tripped on low condenser vacuum, caused by a leaking valve on the isolated pump condenser. Corrective action will consist of the repair of the valve.

The reactor trip from 20 percent power and the resulting transient were normal, and the health and safety of the public were not affected.

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INTRODUCTION: The Unit 2 reactor was manually tripped following a main turbine trip on loss of both main feedwater pumps (FWP). FWP-2A was shutdown for maintenance when FWP-2B tripped on a low FWP-2B condenser vacuum signal. The low vacuum was caused by valve 2CM-158 leaking, allowing a flowpath between the FWP-2A condenser (which was being vented to atmosphere) and the FWP-2B condenser.

Unit 2 was in Mode 1 at 20% power and in the process of shutting down for an outage at the time of the event.

Component Failure/Malfunction, is considered to be the cause of the event, manual valve 2CM-158 did not fully seat when closed.

EVALUATION: FWP-2A was isolated for maintenance as Unit 2 was in the process of shutting down for a short outage. FWP-2A condenser was isolated and its vacuum broken. Operations personnel watched the pressure instrumentation for ∿15 minutes to verify FWP-2A condenser vacuum had dropped to ∿0" Hg and to verify FWP-2B condenser vacuum had not been affected. When vacuum was close to 0" Hg, Operations personnel stopped their surveillance of the pressure instrumentation. The vent on FWP-2A condenser was left open per procedure (Main Vacuum Priming) and, within ∿2 minutes, vacuum started to increase in FWP-2A condenser. Valve 2CM-158 was found to be leaking, creating a flowpath betwen FWP-2A and FWP-2B condensers. The following sequence is believed to have occurred.

While vacuum was decreasing in the FWP-2A condenser, water was leaking through valve 2CM-158 at a slower rate than air was coming into the open vent. This leakage was not sufficient to affect the vacuum in FWP-2B condenser. Approximately two minutes after the vacuum reached 0" Hg in the FWP-2A condenser, it is believed

that the water in the FWP-2A condenser dropped to the level of valve 2CM-158. At this point, air (instead of water) was flowing by the leaking valve. As a result, the vacuum started to increase in FWP-2A condenser and decrease in FWP-2B condenser, as the pressure differential between the condensers diminished.

Main condenser vacuum was not greatly affected by the leak due to the difference between the main condenser and the FWP condenser volumes. A loop seal also prevented air in-leakage to the main condenser. In the 10 minutes prior to FWP-2B trip, main condenser vacuum decreased 0.3" Hg.

When the low vacuum alarm for FWP-2B was received, it was too late to prevent the trip. The alarm setpoint is 20" Hg and the trip setpoint is 14" Hg. Vacuum was decreasing rapidly, so by the time the open vent could have been closed and the FWP condensers reached equal pressure (the leak would have continued as long as a differential pressure existed across 2CM-158), FWP-2A condenser vacuum would have reached its trip setpoint.

CORRECTIVE ACTION: Valve 2CM-158 is a 8" manually operated gate valve and was thought to have been tightly closed while isolating FWP-2A. Apparently the valve seat is damaged, or something is lodged in the seat and prevents full seating. The valve will be repaired or replaced during an upcoming outage.

SAFETY ANALYSIS:

Reactivity was properly controlled by the manual reactor trip Pressurizer pressure was slightly higher than normal pre-trip because two banks of pressurizer heaters were energized. Pressure increased following the turbine trip as primary average temperature rose, and peaked at 2283 psig. Pressure remained below the PORV setpoint of 2335 psig. Pressure dropped immediately following the reactor trip to 2218 psig and slowly decreased, as steam pressure fell, to its minimum value of 2186 psig in about 4 minutes. This is well above the Safety Injection setpoint of 1845 psig. (Steam pressure response will be discussed below.) Pressure recovered to its post-trip maximum value of 2275 psig about 14 minutes after the reactor trip, through the action of the pressurizer heaters. Pressure settled out within 5 psi of its reference value (2235 psig) thirty minutes after the reactor trip. Reactor coolant system wide range hot leg temperature increased ~1°F prior to the trip as main feed-water flow decreased because of increasing condenser back pressure. Temperature increased another 1°F following the turbine trip as the steam dump controller maintained

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

average coolant temperature at the reference temperature plus a 3°F deadband. Temperature decreased after the reactor trip as heat input decreased and steam pressure fell because of the auxiliary feedwater addition. Hot leg temperature stabilized at ~552°F. The expected no-load temperature is 557°F. Wide range cold leg temperature also increased 1°F prior to the turbine trip due to the reduction in feedwater flow. Cold leg temperature increased 4°F following the turbine trip as steam pressure rose. The temperature then turned with steam pressure and stabilized at 552°F. The minimum reactor coolant average temperature was 552°F. Pressurizer level increased 2% prior to the turbine trip as coolant temperature increased. Following the turbine trip, level spiked 3.3% when temperature rose and then decreased, after the reactor trip, to its minimum value of 21.2%. Level gradually recovered to its expected no-load value of 25% about thirty minutes after the turbine trip. Pressurizer level remained on scale at all times. Letdown was not isolated.

Steam pressure was initially ~1015 psig. Prior to the turbine trip, pressure increased to 1035 psig. Pressure increased to 1075 psig after the turbine trip when the condenser dump valves controlled average coolant temperature in the load rejection mode. Following the reactor trip, pressure increased to 1096 psig when the steam dump controller shifted to plant trip mode. The steam generator PORVs and code safety valves were not challenged. (Their setpoints are 1125 psig and 1170 psig respectively.) Pressure then decreased as a result of the high auxiliary feedwater flow rate until the operators throttled flow about four minutes post-trip. Pressure then stabilized at ~1035 psig, and began to recover once auxiliary feedwater flow was reduced below 100 gpm 27 minutes after the trip. Thirty minutes after the trip pressure was 1040 psig and recovering towards its expected no-load value (1090 psig). Steam generator level decreased from 42% to 37% narrow range prior to the turbine trip as main feedwater flow decreased. Main feedwater flow ceased when pump 2B tripped. Main feedwater was isolated about 5 seconds after the reactor trip on reactor trip with coincident low average coolant temperature. Level dropped following the turbine and reactor trips as steam pressure increased and the voids collapsed. Minimum steam generator level was 17% in steam generator A, above the post-trip low-low level setpoint of 12%. The two motor driven auxiliary feedwater pumps initiated on loss of both main feedwater pumps, and fed the steam generators. Level had returned to 27% by four minutes after the turbine trip as a result of the auxiliary feedwater addition. At

that time the operators reset the auxiliary feedwater control valves and reduced the flow. Auxiliary feedwater flow was decreased gradually, and steam generator level recovered smoothly toward its no-load target (38%). Steam generator level was within $\pm$ 5% of the no-load target thirty minutes after the trip. The steam generator level recovery was well controlled by the operators using auxiliary feedwater. Steam pressure remained steady once the auxiliary feedwater control valves had been reset.

Safety Injection was not actuated during this event. The pressurizer PORVs and Code Safety Valves were not challenged. Indicated pressurizer and steam generator level remained on scale. The primary cooldown rate was approximately 17°F per hour, well within the 100°F per hour Technical Specification limit. The Main Steam Code Safety Valves were not challenged. No abnormal release of radioactivity occurred during this event, and there was no abnormal primary leakage.

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

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December 17, 1984

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

Subject: McGuire Nuclear Station, Unit 2
Docket No. 50-370
LER 370/84-29

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 370/84-29 concerning a manual reactor trip resulting from a turbine trip 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 November 15, 1984. 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

Hal B. Tucker

SAG/mjf

Attachment

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cc: Mr. W. T. Orders
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