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 6

TITLE (4)

Reactor Trip on Apparent High Negative Flux Rate

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 2	0 5	8 5	8 5	— 0 0 6	— 0 0	0 3	0 7	8 5		0 5 0 0 0
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THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

OPERATING MODE (9)	1									
POWER LEVEL (10)	0 9 4		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 - 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)*	X NO				

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

On February 5, 1985, McGuire Unit 1 tripped on a high negative flux rate signal. The cause of the trip signal could not be determined, due largely to the inoperability of the events recorder and the relatively coarse data-point intervals of the alarm typer (5 seconds) and the transient monitor (1 second).

A meeting was held to evaluate the trip and to attempt to identify the cause before the decision was made to restart.

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Introduction: On February 5, 1985, at 1320, the Unit 1 reactor tripped on a high negative flux rate signal. The exact cause of the nuclear instrumentation trip signal could not be determined during post-trip investigations.

An independent review was performed by knowledgeable station personnel to determine the cause of the trip prior to a unit restart. The investigation was inconclusive with the available post-trip data.

Unit 1 was in Mode 1 at 94 percent power at the time of this event.

The cause of this event is unknown because the root cause of the negative flux rate trip signal could not be determined.

Evaluation: On February 5, 1985, the Unit 1 reactor tripped from an apparent high negative neutron flux rate signal. The exact cause of the reactor trip could not be determined from the available post-trip data. The event recorder inputs for the reactor trip breaker operating times was inoperable because of a failed input circuit card. The input card was identified as being inoperable, since the previous reactor trip on January 28, but had not been repaired. This card was replaced and tested following this event. Station personnel met to discuss the trip in an attempt to determine the root cause. The factors considered during the investigation were as follows:

1) Three of the four negative rate trip setpoints had been changed, per Westinghouse recommendation, from 5 percent power drop in 2 seconds to 2.5 percent power drop in 2 seconds. The fourth channel setpoint had not been changed at the time of this event. No actual work was in progress when the trip occurred, and all four channels were functional.

2) Personnel were inspecting a wiring modification in the reactor trip breaker cabinets at the time of the event.

3) Operations personnel observed digital rod position indications showing rods dropping into the core just prior to the high negative rate trip alarms.

4) Electrical noise on the power range neutron detectors was investigated.

5) The possibility of multiple dropped rods or control rod drive failures was investigated.

6) The possibility of a reactor trip breaker (RTB) failing open was discussed.

7) Previous reactor trip data was compared to this trip to help determine a probable cause.

The negative rate trip setpoint change was initiated by Westinghouse based upon a corrected interpretation of the safety analysis (see LER 369/85-05). The setpoint change increased the sensitivity of the bistable trip point. Station personnel reviewed the setpoint change as a possible cause of the reactor trip. If noise or voltage spikes had occurred on two of the four power range channels simultaneously, it would be possible for this to have caused the trip. The Unit 1 neutron detectors were found to be relatively free of noise and spikes and this theory was discounted. Recorders were placed on the power range channels to monitor noise and spikes during the subsequent unit startup. There was no evidence of excessive noise or voltage spikes during the period of observation.

Personnel were inspecting a wiring modification in the reactor trip breaker cabinets at the time of the event. The wiring problem under review was a train separation deficiency in which the "Train A" RTB wiring was not properly physically separated from the "Train B" wiring following a modification. The personnel were physically inside the cabinets determining an appropriate routing path for the associated wiring.

Inside the rear of each breaker cabinet are test switches used for time response and shunt trip testing. These switches are labeled "UV Test" (undervoltage test) and "ST Test" (shunt trip test). The UV test switch only disables the shunt trip actuation circuit and would not cause a reactor trip breaker to open. This switch is used during the time response testing of the undervoltage trip circuitry. The ST test switch will actuate the shunt trip circuitry instantly if depressed and trip the associated reactor trip breaker. The personnel were working close to these switches and state that no switches or components were tampered with. The personnel were not aware of the reactor trip until informed by Operations personnel.

Loose wiring connections were found on the 48 VDC terminals in both Train A and B cabinets. The wire connections were loose enough to be physically moved by hand but were still snug enough to provide electrical contact. Instrument and Electrical (IAE) personnel tried moving these wires in an attempt to recreate the reactor trip. It was determined that these loose wires could not have caused the trip. These terminals are used during time response testing and have alligator clip test leads attached to them during each test. The terminals were tightened by Instrument and Electrical personnel following their discovery.

During the event, all four quadrants of the power range channels in the upper and lower levels were basically uniform, indicating that all four channels were responding to the same flux change. This would rule out the possibility of several dropped rods causing the high negative rate signal. It would be unlikely for the outputs of all four detectors to respond equally during this type event. Also if a dropped rod condition exists, an urgent rod control failure alarm would be initiated. This alarm was not received.

Operations personnel state that the digital rod position indicators showed rods falling into the core prior to receiving the high flux rate alarm. This indicates that the reactor trip breaker(s) may have opened prior to the high negative flux rate trip. The one second resolution of the transient monitor trip breaker opening times does not provide proof as to whether a trip breaker opened first, thus causing a negative flux rate trip signal on actual dropping rods.

IAE personnel reviewed previous trip data to determine approximate reactor trip breaker opening times. All the various trip conditions reviewed indicated that the breaker actuation times were less than one second. No conclusions could be made from this review because the time resolution from the transient monitor data was only one second.

Transient Analysis

Reactivity was controlled by the control rod insertion. Pressurizer pressure fell to a minimum of 2030 psig following the trip. Pressure then recovered to its reference value (2235 psig) about 16 minutes later.

Reactor coolant loop average temperature decreased immediately after the trip to ~560°F. Average temperature then trended with steam pressure, reaching a minimum of 554.4°F about 14.5 minutes post-trip. Average temperature stabilized near the no-load target of 557°F 20 minutes after the trip. Reactor coolant loop wide range hot leg and cold leg temperatures tracked together, as designed.

Pressurizer level dropped following the trip to 36%. It then decreased gradually toward its no-load target level (25%) over the next 13 minutes. Minimum level was 19.4%.

Reactor coolant flow responded as expected during this event. No change in reactor coolant pump status occurred.

Steam pressure peaked at 1137 psig in steam generator A. None of the steam generator PORVs opened. The steam pressure decreased over the next three minutes to 1060 psig. At that time, auxiliary feedwater was throttled. Steam pressure then recovered to its no-load target (1092 psig) during the next 12 minutes. Once main feedwater flow to the steam generators was established, pressure stabilized at ~1050 psig. This response was typical for a McGuire trip with good steam generator level and steam pressure control.

Steam generator level dropped immediately after the reactor trip to ~30% narrow range. Main feedwater was isolated about 20 seconds after the trip on reactor trip with coincident low average primary coolant temperature. Both main feedwater pumps tripped shortly afterwards on high discharge pressure. Steam generator level then increased because steam pressure decreased. Auxiliary feedwater flow was throttled once all four steam generator levels were at or above the no-load target (38% narrow range).

The steam generator level then gradually decreased about 8% over the next eight minutes as auxiliary feedwater flow was not quite sufficient to maintain level. Auxiliary feedwater flow was increased about eleven minutes after the reactor trip, and the levels began to increase in steam generators A, B and D. Level in steam generator C remained constant. Auxiliary feedwater flow to steam generator C was approximately 60 gpm lower than in the other steam generators.

The B main feedwater pump was reset about six minutes after the reactor trip. Seven minutes later, the main feedwater pump discharge pressure exceeded the steam generator pressure, and main feedwater was supplied to the steam generators. A sharp jump in steam generator level occurred at this time. About one minute later, the auxiliary feedwater flow to all four steam generators was terminated. Level had recovered above the no-load target in steam generators B, C, and D by twenty-one minutes after the reactor trip.

The valve actuator on 1CF-126 (main feedwater to upper nozzle isolation) was found to be damaged after it had successfully closed on feedwater isolation. This prevented feeding main feedwater to steam generator A through the upper nozzle following the trip, until the valve could be repaired. After auxiliary feedwater was isolated from steam generator A, the level slowly decreased. The operators restored auxiliary feedwater flow to steam generator A about twenty minutes after the trip, and the level began to increase. Level was 30% twenty-one minutes after the reactor trip.

IEMF34 (steam generator C sample high radiation) alarmed following the trip causing blowdown isolation. This was attributed to the suspected very small primary to secondary leak in steam generator C.

Safety Assessment

Reactivity was controlled by the rod insertion. Residual heat was removed by main and auxiliary feedwater to the condenser. No steam was released to the atmosphere. Adequate core cooling was maintained at all times. The primary system remained subcooled. Pressurizer level remained above the low level alarm setpoint. The core was cooled by forced reactor coolant system flow. Pressure remained below the setpoint for the pressurizer PORVs and code safety valves. No abnormal reactor coolant leakage occurred as a result of this event. This event did not impair safety system availability.

The reactor trip breakers were time response tested following this event to verify their mechanical operation. The test data was acceptable and was approximately the same time response as previous test results.

Following the reactor trip, the auxiliary feedwater doghouse header check valve (ICA-49) failed to close as the turbine driven auxiliary feedwater pump (AFWPT) was secured from service. This check valve has stuck open on two previous occasions. These two events are discussed in LER 369/85-04.

Corrective Action:

Prior to Restart: The reactor post-trip review was performed. The cause of the reactor trip could not be determined.

An independent review committee was assembled to investigate the cause of the reactor trip. The evidence produced during this investigation was also inconclusive. All related systems were reviewed and/or tested for satisfactory operation.

ICA-49 (doghouse header check valve) was closed by manual mechanical manipulation. Plans for a permanent repair on this valve are outlined in LER 369/85-04.

Subsequent: Electronic recorders were placed on the Nuclear Instrumentation system to monitor for possible spikes on the neutron flux channels. (No unusual or excessive noise was identified during the subsequent unit restart).

The events recorder points for the reactor trip breaker status were repaired and verified to be operable prior to the unit restart.

Planned: Station personnel are reviewing measures to identify and/or restrict access to cabinets with high trip potential. These measures may include warning signs, Operations or IAE clearance to open cabinets, and possibly, locking devices on cabinet doors.

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March 7, 1985

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

Subject: McGuire Nuclear Station, Unit 1
Docket No. 50-369
LER 369/85-06

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 369/85-06 concerning a Reactor Trip on a High Negative Flux Rate Signal which is submitted in accordance with §50.73 (a)(2)(iv). 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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