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	1 OF 0 3

TITLE (4)
Reactor Trip on Erroneous Signal

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 8	3 1	8 4	8 4	0 2 1	0 0	1 0	0 1	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 0 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
Scott Gewehr	AREA CODE 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)

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 August 31, 1984 McGuire Unit 2 tripped from 100% power on an inadvertant 2 out of 4 channel power range high flux rate signal. The signal was generated during performance of a test procedure as one channel of the circuit was taken out of service for testing, and a power supply lead in a second channel was mistakenly lifted, resulting in the 2 out of 4 logic trip.

Personnel error is considered to have been the major cause of the event. All plant systems responded as intended following the trip. Corrective actions include counseling and instruction to appropriate personnel to avoid similar errors of this nature in the future, procedural enhancements which recognize, and thereby guard against, the potential for such errors, and improved labeling of nuclear instrumentation cabinets.

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

INTRODUCTION: On August 31, 1984, the unit two reactor tripped on a two out of four power range (P/R) high flux rate signal. This signal was generated inadvertently by Instrument and Electrical (IAE) personnel while performing a test procedure (Nuclear Instrumentation System (NIS) Power Range Rate Circuit and Bistable Relay Drivers Alignment (EIIS SYSTEM CODE: JC)).

P/R channel 43 was placed in the trip mode in preparation for testing. The power supply cable for P/R channel 42 was then mistakenly unplugged (instead of P/R channel 43), placing P/R channel 42 in the trip mode also. With two P/R channels (out of a total four channels) in the trip mode, a reactor trip was automatically initiated.

Personnel error is considered to be the major cause of the event. However, procedural enhancements and hardware additions (e.g. Labels for Cabinets) have been identified which should minimize the probability of occurrence.

EVALUATION: The test procedure must be completed every 18 months to meet the surveillance requirement of Technical Specification 4.3.1.1 (channel calibration of P/R neutron flux setpoints). This test is completed on one P/R channel at a time, using a generic procedure with procedure steps that apply to any of the four P/R channels. The steps do not refer to the channel being tested; therefore, the IAE technician performing the test must keep in mind which channel is being tested.

On the day of the event, IAE technician A removed the instrument fuses on the front of the N/I cabinet for P/R channel 43. IAE technician A walked around a row of cabinets to get to the back of the N/I cabinet containing P/R channel 43 to disconnect channel 43's input plugs. (IAE technician B, who was assisting with the test, stayed at the front of the cabinets). IAE technician A opened the cabinet door for P/R channel 42 instead of the door for channel 43, and disconnected the input plugs on channel 42. This now placed both P/R channels 43 and 42 in the trip mode. With two P/R channels in the trip mode, a reactor trip was initiated.

The label for P/R channel 43 is on a column between the cabinet doors for channel 43 and 42. Had the label been on the door itself, it may have caught the technicians attention and helped him realize that he was opening the wrong door. There are no labels inside the cabinet to identify the instrumentation contained within. Once the incorrect door was opened, it was unlikely that the technician would have realized he was working on the wrong channel.

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

Reactivity was promptly controlled by the reactor trip as the control rods inserted. Pressurizer pressure responded as expected, dropping to a minimum of 2015 psig before recovering and stabilizing at its reference value of 2235 psig. The pressurizer PORV's and Code Safety Valves were not challenged. Reactor coolant loop average temperature responded as desired, dropping to ~560°F and stabilizing there. Temperature decreased slightly to ~559°F about 30 minutes after the reactor trip. This is slightly above the expected no-load value of 557°F. Wide range hot leg and cold leg temperatures also responded as designed. Pressurizer level control was normal; level dropped immediately after the trip to ~37%, and slowly decreased toward its no-load value of 25%. The pressurizer level stabilized at 25% about 30 minutes after the reactor trip.

Steam pressure peaked at 1132 psig, and stabilized at 1095 psig. This is within 3 psi of its no-load target (1092 psig). The Main Steam Code Relief Valves (setpoint 1170 psig) were not challenged. Steam generator level dropped immediately following the trip to the minimum level of 28% narrow range. Main feedwater was isolated shortly after the reactor trip on reactor trip with coincident low average primary coolant temperature. Both main feedwater pumps tripped on high discharge pressure following the main feedwater isolation. All three auxiliary feedwater pumps were actuated immediately after the reactor trip on indicated low-low steam generator level, and were used by the operators to recover level. Auxiliary feedwater was secured within 24 minutes after initiation as one main feedwater pump was returned to service. Main feedwater was subsequently used to maintain the steam generator levels. The levels were well controlled at all times. Level remained well above the post-trip low-low level setpoint of 12% narrow range.

Safety Injection was not actuated during this event. The pressurizer PORV's and Code Safety Valves were not challenged. Indicated pressurizer and steam generator levels remained on scale. The primary cooldown rate was approximately 30°F/hour, well below the Technical Specification limit of 100°F/hour. No abnormal release of radioactivity occurred during this event, and there was no abnormal primary leakage.

CORRECTIVE ACTION

Appropriate individuals have reviewed the incident and have been made aware of techniques to reduce the liklihood of recurrence of similar events. An evaluation will be performed by November 1, 1984 to identify appropriate procedural improvements.

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

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October 1, 1984

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

Subject: McGuire Nuclear Station, Unit 2
Docket No. 50-370
LER 370/84-21

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 370/84-21 concerning a reactor trip resulting from an erroneous signal, 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 August 31, 1984. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

H.B. Tucker
Hal B. Tucker

SAG/mjf

Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
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