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)

Actuation of Reactor

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

YES (If yes, complete EXPECTED SUBMISSION DATE) NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

On December 21, 1985, Unit 2 tripped from 100% power when preventive maintenance on Unit 1 125 VAC Vital Instrument and Control Power resulted in a Unit 2 inverter being erroneously removed from service, instead of the adjacent Unit 1 inverter. The resulting loss of power to the analog controllers for steam generator (S/G) level, feedwater flow, and steam flow resulted in a feedwater transient, which was corrected by switching to manual control and transferring the controllers to another channel. However, the transfer to another channel was done incompletely, in that one of thirteen controller switches was not moved to the alternate channel. When control was returned to the automatic mode, the contribution of this inoperable input to automatic control caused the level in S/G C to fall to the low-low trip setpoint.

The cause of the event was personnel error because an operator and an independent verifier failed to identify properly the equipment to be removed from service. In addition, the transfer of the steam generator program to an alternate channel was performed incorrectly.

Corrective action will include a re-emphasis with operators on the importance of following procedures, and verification. Also, the S/G low-low level trip setpoint will be lowered to allow operators more time to diagnose and compensate for S/G level transients.

This event had no impact on the health and safety of the public.

IE22 1/1

Introduction: On December 21, 1984 Operations personnel were in the process of shutting down the 120 VAC, Channel II, inverter 1EVIB for preventive maintenance. The operators had correctly applied the alternate supply power to the 1EVIB manual transfer switch which would have supplied power to the vital loads on bus 1EKVB when 1EVIB was removed from service. The operators mistakenly actuated the 2EVIB manual transfer switch for bus 2EKVB, removing it from service without having the Unit 2 alternate supply power available to supply the Channel II instrument and control loads.

The Unit 2 analog controllers for steam generator level, feedwater flow, and steam flow were taking input from the Channel II instruments at the time power was lost to them. The power loss to Channel II caused the Reactor Protection Bistables for that channel to fail to the "trip" condition. The feedwater transient following the loss of power to the controlling instrumentation was corrected by swapping to manual control and switching the controllers to another channel. The level program selector for the nuclear power contribution was left in the N41-N42 position with N42 being inoperable due to the Channel II power loss. When the Control Operators thought steam generator level was under control, they placed the controls in automatic, which caused the levels in B and C to drop. The C steam generator level dropped to the low-low level trip setpoint at 0629:20 causing the reactor trip.

The cause of this event was personnel error, due to the failure of operator A and independent verifier, operator B, to properly identify the equipment being removed from service and failure to follow procedure steps. A contributing factor for the unsuccessful recovery from this plant transient was the incomplete swapping of the steam generator program selector switch for N42.

Unit 2 was in Mode 1 at 100% power at the time of the event.

EVALUATION: The 120 VAC vital instrumentation and control power system provides a source of continuous power for safety related circuits required for operation of each unit. The system consists of four DC/AC inverters per unit supplied with 125 VDC from the charger/battery for the associated channel. A regulated AC power supply (1KRP, 2KRP) is provided for each unit as an alternate source for the AC vital loads. This allows an uninterruptable manual transfer of power when an inverter must be taken out of service.

Nuclear Control Operator A and Nuclear Equipment Operator B were assigned to remove 1EVIB and 2EVIB from service to facilitate preventive maintenance on the inverters. Operator A was experienced with this equipment manipulation but Operator B was not. Operator A planned the task and met Operator B in the battery room area.

Arriving at the alternate source power supply for Unit 1 (1KRP), Operator A closed the breaker that provided an alternate AC source to the manual transfer switch for 1EVIB. Inverters 1EVIB and 2EVIB are located beside each other and encircled by other power panels. The inverters are clearly labeled, 1EVIB and 2EVIB, with the manual transfer switch mounted on the side of each respective cabinet.

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 3 4	0 0	0 3 OF 5

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

Operators A & B went from 1KRP to 2EVIB inverter and turned the Manual Bypass Switch to "Alternate Source to Load" position. Operator A had checked the "IN SYNC" indicator lamps on the inverter and transfer switch (one light on each) and presumed them to be burned out since they were not illuminated.

The procedure used by the operators was 125 VDC/120 VAC Instrument and Control Power / Inverter shutdown. This is a generic procedure used for all eight inverters on Unit 1 and Unit 2. The procedure is adequate to perform the inverter shutdown task but not specific as to component identification and independent verification.

The removal of inverter 2EVIB from service without an alternate AC power supply resulted in loss of 120 VAC to all the Channel II vital instrumentation and controls. The Unit 2 steam generator level control system was set to the Channel II instrumentation to control the steam generator levels.

The Steam Generator Program Level Control System is equipped to maintain a programmed water level which is a function of nuclear power. The system controls the feedwater regulator valve by continuously comparing the selected steam generator level signal, nuclear power and level program signals, feedwater flow signal, and the pressure compensated steam flow signal. A signal also controls the main feedwater pump speed to maintain a programmed differential pressure between the steam header and the feedwater pump discharge pressure. Manual control of the feedwater control valve is available at all times by use of the manual/auto station.

The purpose of the steam generator level program is to determine how much mass will be in the steam generator at different power levels. The level should neither fall to a level that would uncover the tubes nor increase to a level that would flood the moisture separators.

The loss of power to the selected steam generator level control channels caused the four feedwater control valves to go the full open position producing a feedwater swing. Control Room Operator C and Operator D immediately switched from Channel II to Channel I controls for all the selector switches except the "nuclear power" switch. Operators C and D placed the manual/auto station into "manual" to stabilize the steam generator levels. As the levels stabilized, the control operators placed the manual/auto stations back into "auto", believing that the levels were being controlled satisfactorily. The automatic level program circuit was still receiving a zero power signal from the selected, inoperable N-42 channel which was causing the program levels for steam generators B and C to decrease to the 38% level. The level in steam generator C decreased to the 55% trip point causing a reactor trip.

The nuclear power selector switch is labeled N41-N42 / N43-N44. The other 12 selector switches are labeled Channel I / Channel II. It was not readily apparent to the control room operators that N42 selector switch was channel-related and contributing to the steam generator level decrease. Operator action in regard to the Nuclear Power Selector Switch is descibed in the procedure titled Malfunction of Nuclear Instrumentation System.

Operators A and B closed the alternate supply breaker for 2EVIB (2KRP) approximately 8 minutes after the reactor trip, restoring AC power to the Channel II circuits. Inverter 1EVIB was removed from service as originally planned.

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

Pre-Trip Transient

There were some slight changes in major plant parameters pre-trip because of the feedwater flow transient. Initially (on loss of the inverter) all four main feedwater regulating valves opened fully. This increased level in the steam generators, and the increased heat transfer increased steam flow and decreased steam pressure. The increased heat transfer caused average coolant temperature, pressurizer pressure and pressurizer level to decrease slightly also. The moderator temperature effect caused reactor power to increase about 1 percent. Once the steam generator level controls were taken into manual and swapped, average temperature in loops A and B recovered, along with pressurizer pressure and level. Temperature in loop D decreased slightly and then stabilized. At the time of the trip most parameters except steam generator level were near their normal full load values. Peak pressurizer pressure prior to the trip was 2244 psig. Pressurizer level was 63 percent and average temperature was about 586 degrees F at the time of the reactor trip. However, letdown was isolated when the Channel II pressurizer level indication failed to zero.

Post-Trip Response

Reactivity was properly controlled by the reactor trip. Pressurizer pressure dropped to a minimum of 1995 psig about four minutes after the reactor trip before recovering to its reference value (2235 psig) about thirty minutes after the trip. Pressurizer pressure and temperature recovered after the inverter was reenergized and the low pressurizer level alarm and associated heater cutoff was cleared. Pressure remained below the uncompensated PORV setpoint (2335 psig) and above the Safety Injection setpoint (1845 psig) at all times.

Reactor coolant loop average temperature dropped immediately after the trip to 563 degrees F, and declined slowly to 543 degrees F about 22 minutes after the trip. Steam pressure was decreasing at the time, causing the lower than normal primary average temperatures. (The steam pressure response will be discussed in detail below.) Minimum reactor coolant average temperature was 542 degrees F thirty minutes after the reactor trip. This is considerably below the no-load target temperature of 557 degrees F, but the operators were monitoring the temperatures to ensure stability. Reactor coolant loop differential temperature dropped immediately to less than five degrees, and was below two degrees within two minutes after the trip. Wide range hot leg temperature responded similar to average coolant temperature after the trip. The cooldown rate was 46 degrees F per hour, within the Technical Specification limit of 100 degrees F per hour. As peak hot leg temperature was 618 degrees F and minimum pressurizer water temperature was 636 degrees F, the reactor coolant system was subcooled at all times.

Pressurizer level decreased immediately after the reactor trip to about 42 percent and slowly declined to about 26 percent and stabilized (25 percent is the no-load target). Letdown flow was isolated when pressurizer level Channel II deenergized and failed to zero. Letdown was restored about 18 minutes after the reactor trip. Reactor coolant flow was not affected by this event. There was no change in reactor coolant pump status.

Steam pressure peaked at 1117 psig following the trip. Pressure then steadily decreased to about 960 psig about eleven minutes after the reactor trip. At this time, auxiliary feedwater flow to all four steam generators was throttled below 175 gpm, halting the pressure decline. Pressure increased 10 psi over the next seven minutes, and then gradually decreased to 954 psig twenty-nine minutes after the reactor trip. This is significantly below the no-load target of 1090 psig. The reasons for the lower than normal steam pressure are the extended auxiliary feedwater addition and higher than normal steam loads. At the time of the reactor trip, Unit 1 was heating up and had reached 557 degrees F with one main feedwater pump in operation following a four week maintenance outage. Unit 2, along with one auxiliary boiler, was carrying the steam loads of both units. It was decided to allow the existing steam loads on Unit 2 to remain as steam pressure and average temperature had stabilized, even though at lower than normal values.

Steam generator levels dropped sharply after the trip. Minimum level in steam generator A was 10 percent; in steam generator B the level dipped to 28 percent. Level in steam generator C went offscale low for about two minutes following the trip. Steam generator D transient monitor indication was unavailable due to the loss of Channel II. Following the reactor trip, feedwater was isolated and the main feedwater pumps tripped on indicated high-high steam generator level (setpoint 82%) in steam generator D. All three auxiliary feedwater pumps initiated on low-low steam generator level and fed the steam generators. Level recovered quickly with auxiliary feedwater. Level was within 8% of the no-load target in all four steam generators within ten minutes after the reactor trip. The operators reset the auxiliary feedwater control valves and throttled the flow beginning with steam generator D about four minutes after the reactor trip. Flow to the other generators was throttled about eleven minutes after the reactor trip. At that time flow to all four generators was reduced below 175 gpm and steam pressure stabilized.

Safety Function Assessment

The rod insertion controlled reactivity. The residual heat was removed to the condenser by auxiliary feedwater. Adequate core cooling was maintained at all times. The reactor pressure boundary remained intact. Safety Injection was not actuated. The primary and secondary PORVs were not actuated during this event. There was no release of steam to the atmosphere, and no unusual release of radioactivity.

Corrective Action

The importance of following procedures and verifying actions by the use of indicator lights were appropriate, will be re-emphasized to all operating shift personnel.

Labeling on the nuclear power selector switch will be reviewed to determine if channel markings can be added to improve clarity.

A change (decrease) to the Low-Low S/G Level Trip setpoint is being implemented as part of the upcoming refueling outage. This will provide additional time for operator response to feedwater transients.

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

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

January 22, 1985

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

Subject: McGuire Nuclear Station, Unit 2
Docket No. 50-370
LER 370/84-34

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 370/84-34 concerning a reactor trip on loss of vital instrument and control power, 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

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

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

M & M Nuclear Consultants
1221 Avenue of the Americas
New York, New York 10020

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