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 0 8

TITLE (4) REACTOR TRIP DUE TO ERROR ON A SCHEMATIC DIAGRAM WHICH DIRECTED PERSONNEL TO WRONG CABINET - DESIGN DEFICIENCY

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 8	8 7	8 7	— 0 3 6 —	0 0	0 1	2 7	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	0 9 6

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	
STEVEN E. LeROY, LICENSING	7 0 4	3 7 3 - 6 2 3 3

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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 12/28/87 at 1322, during periodic testing, Unit 1 Reactor Trip occurred when Electrical Maintenance (EM) personnel were directed to the wrong Reactor Protective System (7300 RPS) cabinet by an error in a schematic diagram. The error in the schematic diagram caused the personnel to complete a 2 out of 4 logic Reactor Trip signal. A Low-Low Level in Steam Generator 1B did not actually exist. Operations implemented the Reactor Trip Recovery Procedure to recover from the transient. At 2214, EM initiated a Reactor Trip signal when they did not follow procedure and caused a general warning on Train A of the Solid State Protection System (SSPS) while a general warning existed on Train B of the SSPS. Unit 1 was returned to Mode 1, Power Operation, on December 29, at 1249. The Reactor Trip is assigned a cause of Design Deficiency because the schematic diagram was incorrect. Also Personnel Error because personnel did not pursue adequate checks on the diagram. The second Reactor Trip signal is assigned a Personnel Error, because personnel involved did not follow procedure. Also Defective Procedure because no precautions were in the procedure to prevent events of this type. The diagrams were marked as incorrect and changes to correct Units 1 and 2 diagrams were implemented. All other 7300 diagrams were reviewed.

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INTRODUCTION:

On December 28, 1987 at 1322, during periodic testing, a Unit 1 Reactor Trip occurred when Instrumentation and Electrical (IAE) personnel were directed to the wrong Reactor [EIIS:RCT] Protective System (7300 RPS) [EIIS:JC] cabinet [EIIS:CAB] by an error in a schematic diagram. The error in the schematic diagram caused IAE personnel to adjust Channel 4 of the Steam Generator [EIIS:SG] 1B Low-Low Level instrumentation loop which he thought was Channel 2. Since Channel 2 was in a [EIIS:TRB] tripped condition, adjusting Channel 4 completed a two out of four logic Reactor Trip signal. A Low-Low Level in Steam Generator 1B did not actually exist. The Main Turbine tripped because of the Reactor Trip. Operations implemented the Reactor Trip Recovery Procedure to recover from the transient.

At 2214, while performing pre-startup periodic testing on the Nuclear Instrumentation (NI) system [EIIS:BQ], IAE personnel initiated a Reactor Trip signal when they did not follow procedure and caused a general warning on Train A of the Solid State Protection System (SSPS) [EIIS:JC] while a general warning existed on Train B of the SSPS. Unit 1 was returned to Mode 1, Power Operation, on December 29, at 1249.

Unit 1 was in Mode 1, Power Operation, at 96% power, at the time of the Reactor Trip. Unit 1 was in Mode 3, Hot Standby, when the second Reactor Trip signal was generated.

The Reactor Trip has been assigned a cause of Design Deficiency because the schematic diagram for the Steam Generator 1B Low-Low Level instrumentation, Channel 2, was incorrect. A contributing cause of Personnel Error has been assigned to this event because the IAE personnel involved did not pursue adequate checks when the schematic diagram did not appear to be correct.

The second Reactor Trip signal has been assigned a cause of Personnel Error because the IAE personnel involved did not follow procedure. A contributing cause of Defective Procedure has been assigned to this event because there were no precautions in the IAE and/or Operations procedure to prevent events of this nature.

EVALUATION:

Background

The 7300 RPS automatically keeps surveillance on process variables which are directly related to equipment mechanical limitations (e.g. pressures, water levels, etc.), and also on variables which directly affect the heat transfer capability of the Reactor (e.g., flows, temperatures, etc.). Other parameters used in the 7300 RPS are calculated from various process variables. Whenever a direct process or calculated variable exceeds a setpoint, the Reactor is shut down in order to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of fission products into the Containment

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

Building [EIIS:NH]. The 7300 RPS consists of four protection cabinets; each cabinet contains one protection channel. Each channel is completely independent; channels do not cross between cabinets. A functional test is performed monthly on each channel of the 7300 RPS and each channel test takes several days.

Each S/G is provided with four level transmitters [EIIS:LT]. Each transmitter is related to one channel of the 7300 RPS. Any two Low-Low level signals from any one S/G will initiate a Reactor Trip signal (two out of four logic).

The SSPS consists of two redundant logic trains (Train A and Train B) which receive inputs from the 7300 RPS channels and NI channels to complete the logic necessary to automatically open the Reactor Trip breakers. Each train contains a Multiplexer Test switch [EIIS:HS]. At the start of a NI system test, this switch (in either train) is placed in the A+B position. The A+B position allows comparison of Train A and Train B logic so that one channel that inputs to both trains can be tested. When rotating the Multiplexer Test switch from or to the Normal position it has to pass through the Inhibit position. When the switch is rotated through the Inhibit position, a general warning signal is generated; a general warning signal on both trains of the SSPS at the same time will cause a Reactor Trip. A general warning signal will exist if the switch is rotated quickly, even though the general warning lamp [EIIS:IL] does not energize.

Description of Event

On December 28, 1987, at approximately 0800, IAE personnel resumed routine testing on Channel 2 of the 7300 RPS cabinets. The Channel 2 S/G 1B Low-Low Level instrumentation loop was found to be out of tolerance. An error on a schematic diagram led the IAE personnel involved to a Channel 4 circuit card, and when they attempted to make adjustments on the circuit card, a spike resulted in Channel 4 of the S/G 1B Low-Low Level instrumentation loop. Since Channel 2 was in test (tripped condition), a two out of four logic generated a Reactor Trip signal at 1322:20, and the Unit 1 Reactor tripped. The Reactor Trip caused a Turbine Trip immediately thereafter. OPS implemented the Reactor Trip Recovery Procedure to recover from the transient.

On December 28, 1987, at approximately 2214, IAE and OPS were conducting pre-startup testing on NI Channel N-31 and the Manual Reactor Trip system simultaneously. OPS racked in and closed Reactor Trip Breaker BYB [EIIS:BKR] which caused a general warning on SSPS Train B. IAE completed testing on NI Channel N-31 and returned the Multiplexer Test switch on SSPS Train A to Normal which generated a general warning on SSPS Train A. A Reactor Trip signal was generated at 2214 because of the general warning on both trains of the SSPS. A Main Feedwater (CF) system [EIIS:SJ] isolation occurred and was reset by OPS.

Unit 1 entered Mode 1 at 1249 on December 29, 1987.

Conclusion

IAE Specialist A and IAE Technician A were performing monthly periodic maintenance, Protective System Channel 2 Functional Test, on Channel 2 of the S/G 1B Low-Low Level instrumentation loop. They found an out of tolerance reading and determined that the lead/lag circuit card needed to be adjusted. They consulted the schematic diagram series referenced in the procedure to determine the number of the circuit card to adjust. Schematic diagram, Steam Generator Level #1, listed the lead/lag card as C4-326 (cabinet 4, slot 326). The IAE personnel involved thought that the number did not appear to be correct since all the other circuit card numbers on the schematic drawing began with C2. They looked in Cabinet 2 at slot number 326 and found that slot empty. They got permission from the Control Room personnel to open a second 7300 RPS cabinet to locate the lead/lag circuit card listed on the schematic diagram. Slot 326 in cabinet 4 contained a new lead/lag circuit card and the IAE personnel involved assumed that a Nuclear Station Modification (NSM) had relocated the Channel 2 card to cabinet 4. They attempted to adjust the lead/lag circuit card output and caused a spike which indicated a Low-Low Level in S/G 1B on Channel 4 of the 7300 RPS. Since S/G 1B Low-Low Level Channel 2 was in test, a 2 out of 4 logic initiated a Reactor Trip. The root cause of this Reactor Trip was the error in the schematic diagram; therefore, a cause of Design Deficiency has been assigned to this event.

Duke Design Engineering (DE) conducted an investigation to determine when the error was made on the schematic diagram. NSMs were implemented during the 1985 Unit 1 and Unit 2 refueling outages which added lead/lag circuit cards to the S/G Level instrumentation loops. The error was made on Westinghouse drawing number 7389D48 which is for Channel 2 of S/G 1B Level. The error indicated the location for the lead/lag circuit card as Channel 4 of S/G 1B Level which is Westinghouse drawing number 7389D84. DE personnel believe that in the process of upgrading the drawings, Westinghouse drawing number 48 was changed instead of Westinghouse drawing number 84. The Westinghouse drawings in the field package for the NSM were correct. The Unit 2 7300 RPS is identical to the Unit 1 7300 RPS; therefore, the Unit 2 drawing was copied from the Unit 1 drawing and the error was also transferred to the Unit 2 drawing.

A contributory cause of Personnel Error has also been assigned to this event because the IAE personnel involved questioned among themselves the correctness of the schematic diagram but did not pursue it far enough. Although the drawing did not appear correct, they proceeded without contacting IAE Staff personnel. The IAE personnel involved were Employee Training and Qualification System qualified; both had been to school provided by Westinghouse on the 7300 RPS; IAE Specialist A has approximately 2 years experience and IAE Technician A has approximately 4 years experience working on the 7300 RPS.

The second Reactor Trip signal was initiated during independent pre-startup testing being conducted by IAE and OPS. On December 28, at 2130, IAE Specialist B, with IAE Specialist A acting as independent verifier, started pre-startup testing on NI Channel N-31 using period test procedure, Nuclear Instrumentation

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

System Source Range Functional Test. At the start of the procedure, IAE personnel are directed to verify that the general warning lamps on SSPS Train A and Train B are operable and extinguished prior to placing the SSPS Train A Multiplexer Test switch in the A+B position. IAE personnel rotated the Multiplexer Test switch to the A+B position. In the meantime OPS started to perform pre-startup testing on the Manual Reactor Trip system using periodic test procedure, Manual Reactor Trip Function Test. OPS personnel did not expect the two tests to impact each other. OPS racked in Reactor Trip Bypass Breaker BYB which energized the SSPS Train B general warning circuit. The IAE personnel involved completed the functional test on NI Channel N-31 and returned to the SSPS cabinets to return the Train A Multiplexer Test switch to the Normal position as directed by the procedure. The procedure directs IAE to verify that the general warning lamps are extinguished on SSPS Train A and B prior to returning the switch to the Normal position.

When the IAE personnel involved went to the SSPS Train B cabinet, they found the general warning lamp on. They talked to the OPS Assistant Shift Supervisor and found out that Reactor Trip Bypass Breaker BYB was racked in which gave the Train B general warning lamp. The IAE personnel involved and the OPS Assistant Shift Supervisor discussed the situation but specific mention of a Reactor Trip signal was not made; the OPS Assistant Shift Supervisor then said he had no problem with returning the switch to the Normal position.

IAE Specialist B is not trained on the SSPS and he did not know that two general warnings would generate a Reactor Trip signal. IAE Specialist B also thought that rotating the Multiplexer Test switch rapidly would prevent a general warning signal. IAE Specialist A is trained on the SSPS and knew that two general warnings would generate a Reactor Trip signal, but he also thought that rapidly rotating the switch when going from the A+B position to the Normal position would not produce a general warning signal. The IAE personnel involved did rotate the switch to the Normal position rapidly enough not to illuminate the general warning lamp, but the general warning circuitry still saw the signal and generated a Reactor Trip signal.

The procedure, Nuclear Instrumentation System Source Range Functional Test, did contain a step to verify that the general warning lamps were extinguished prior to rotating the Multiplexer Test switch to the Normal position. When the IAE personnel involved found the general warning lamp on Train B illuminated, they only checked with OPS instead of checking with the appropriate IAE Staff personnel prior to rotating the switch. They also should have waited for OPS to complete the Manual Reactor Trip Function Test prior to rotating the switch. Therefore, this event has been assigned a cause of Personnel Error.

A contributory cause of Defective Procedure has also been assigned to this event. Since the procedure, Nuclear Instrumentation System Source Range Functional Test, is normally performed by personnel trained on the NI system who are not trained on the SSPS, a caution should be in the procedure relating to the SSPS Multiplexer Test switch operation. The OPS procedure, Manual Reactor Trip Function Test, should also include a prerequisite relating to other train testing in progress.

Several anomalies occurred during this Reactor Trip. S/G D Containment Isolation Main Feedwater valve [EIIS:ISV], valve 1CF-26, opened after OPS manually closed the valve. OPS wrote an emergency priority work request to investigate/repair valve 1CF-26. OPS personnel said that problems with the S/G Containment Isolation Main Feedwater Valves are recurring. IAE has identified the problem as being associated with the valve accumulator pressure switches [EIIS:PS]. The pressure switches are the wrong type for the application and will not maintain the proper setpoints. Turbine Driven CF Pumps [EIIS:P] 1A and 1B tripped on high discharge pressure, OPS and IAE investigated and did not find any problems. (The same problem occurred on a subsequent Reactor Trip on January 7, 1988. IAE personnel investigated and corrected the problem.) The new Events Recorder is not provided with as many inputs as the old Events Recorder to accurately determine a sequence of events on a Reactor Trip. Performance personnel are pursuing corrective action to reconnect the deleted inputs to the new Events Recorder.

During the Reactor Trip, OPS responded to the transient in a timely manner to stabilize the unit. All primary and secondary key parameters were at approximate no-load conditions thirty minutes after the trip. The open setpoints for all four S/G Power Operated Relief Valves (PORVs) [EIIS:RV] were exceeded and all four valves opened. S/G 1B PORV, valve 1SV-13, opened early and closed late. Performance (PRF) wrote a routine priority work request to investigate and repair valve 1SV-13. IAE personnel believe that one problem associated with valve 1SV-13 opening early and closing late is the timing of the data retrieval during a Reactor Trip. The maximum S/G pressure reached was approximately 1135 PSIG, which is below the open setpoints of the Main Steam [EIIS:SB] Line Code Safety Valves [EIIS:RV], and they did not open. Pressurizer (PZR) [EIIS:PZR] pressure reached approximately 2270 PSIG, which is below the open setpoints of the PZR PORVs [EIIS:RV] and PZR Code Safety Valves [EIIS:RV], and the valves did not open. CF isolation occurred on a Reactor Trip with Low T-ave. The S/G 1B Low-Low Level signal actuated the Auxiliary Feedwater (CA) system [EIIS:BA] and the Motor [EIIS:MO] Driven CA pumps [EIIS:P] started to supply feedwater to the S/Gs.

The Steam Dump To Condenser Valves [EIIS:FCV] responded properly and modulated as necessary to provide a heat sink for the Reactor. Steam Dump To Condenser Valves 1SB-3 and 1SB-6 had been isolated due to seat leakage and did not operate. There are work requests outstanding to repair valves 1SB-3 and 1SB-6.

On December 29, at 0153, while Unit 1 was in Mode 3, Hot Standby, "S/G Flow Mismatch Low CF Flow Alert" alarms were received in the Control Room. Steam header pressure indicated high and the Steam Dump To Condenser Valves opened. Reactor Coolant T-ave dropped from 557 degrees-F to 540 degrees-F, and PZR level dropped to approximately 15 percent. Loss of letdown resulted from the low PZR level. OPS implemented the Loss Of Letdown, Charging, or Injection procedure to restore normal letdown and charging. The cause of this transient was water in a steam pressure transmitter which was caused by steam from a steam leak blowing on the transmitter. The steam leak was isolated, and the transmitter was dried out and returned to service.

A review of the McGuire Reports revealed numerous Reactor Trips. Licensee Event Reports (LER) 369/87-21, 370/86-16, 370/86-21, and 370/85-26 described Reactor Trips caused by personnel error. LERs 370/86-02 and 370/85-13 described Reactor Trips caused by procedure deficiencies. Therefore, this event is considered recurring. The corrective actions for these events were directed at specific procedures or components and would not have prevented this event.

This event is not reportable to the Nuclear Plant Reliability Data System (NPRDS).

CORRECTIVE ACTIONS:

Immediate:

1) OPS implemented the Reactor Trip Recovery Procedure.

2) OPS reset the CF isolation that occurred when the second Reactor Trip signal was received.

3) IAE Staff personnel marked the Unit 1 and Unit 2 Channel 2 S/G 1B and 2B Level instrumentation loop schematic diagrams as being incorrect.

Subsequent:

1) DE implemented changes to correct the Unit 1 and Unit 2 S/G 1B and 2B Level instrumentation loop schematic diagrams.

2) DE reviewed all of the 7300 RPS schematic diagrams involved in the 1985 NSM and did not find any additional errors.

Planned:

1) IAE will follow up with PRF on the problem with valve 1SV-13 to upgrade data retrieval and/or analysis to more accurately determine PORV lift times.

2) The replacement of the accumulator pressure switches for the S/G Containment Isolation Main Feedwater Valves will be evaluated by a Station Problem Report that IAE will initiate.

3) IAE will initiate short term training and communication on the 7300 RPS cabinet door control and provide input to Production Training Services, OPS, and PRF.

4) IAE will initiate a discussion with appropriate IAE, OPS, and PRF personnel regarding SSPS general warnings and each group will provide administrative controls (training, communications, and procedures) as necessary to prevent incidents pertaining to the SSPS general warning alarms.

SAFETY ANALYSIS:

This Reactor Trip was caused by a S/G 1B Low-Low Level signal. A Low-Low Level in S/G 1B did not actually exist. The unit responded to the Reactor Trip without any significant problems. The key primary and secondary parameters were at their approximate no-load value 30 minutes after the trip. Adequate core cooling was maintained throughout the transient, and the NC system pressure boundary was not challenged. The Steam Dump To Condenser Valves opened to remove heat from the Reactor. The open setpoints for the S/G PORVs were exceeded, and the valves opened to reduce S/G pressure. The open setpoints for the S/G Code Safety Valves, PZR PORVs, and the PZR Code Safety Valves were not reached and the valves were not challenged. Emergency power and/or emergency core cooling were not required in this event and they were not actuated.

There were no personnel injuries, personnel overexposures, or releases of radioactive material as a result of this event.

This event is considered to be of no significance with respect to the health and safety of the public.

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

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

January 27, 1988

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

Subject: McGuire Nuclear Station, Unit 1
Docket No. 50-369
Licensee Event Report 369/87-36

Gentlemen:

Pursuant to 10CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 369/87-36 concerning a reactor trip that occurred on December 28, 1987. This report is being submitted in accordance with 10CFR 50.73(a)(2)(iv). This event is 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

SEL/213/jgc

Attachment

xc: Dr. J. Nelson Grace
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, GA 30323

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

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

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

Mr. Darl Hood
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

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

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