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0273I
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May 16, 1988

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

PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
EQUIPMENT FAILURE IN CONJUNCTION
WITH SURVEILLANCE CAUSES SCRAM

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning the unanticipated actuation of some Engineered Safety Features (ESFs). The event occurred at Plant Hatch - Unit 2.

Sincerely,

R. P. McDonald

CLT/ct

Enclosure: LER 50-366/1988-011

c: (see next page)

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Georgia Power

U. S. Nuclear Regulatory Commission
May 16, 1988
Page Two

c: Georgia Power Company
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Mr. L. T. Gucwa, Manager Nuclear Safety and Licensing
GO-NORMS

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0273I

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

# LICENSEE EVENT REPORT (LER)

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TITLE (4)
EQUIPMENT FAILURE IN CONJUNCTION WITH SURVEILLANCE CAUSES SCRAM

| 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 4 | 1 7 | 8 8 | 8 8 | — 0 1 1 — | 0 0 | 0 5 | 1 6 | 8 8 | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): 1
POWER LEVEL (10): 0 8 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) |
|---|---|
| J. D. Heidt, Nuclear Licensing Manager - Hatch | 4 0 4 526-4630 |

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 |
|---|---|---|---|---|---|---|---|---|---|
| X | I G | R J X | G 0 8 0 | Y | X | I D | E C B D | G 0 8 0 | N |
| | | | | | | | | | |

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 04/17/88 at approximately 0140 CDT, Unit 2 was in the run mode at an approximate power level of 1949 MWt (approximately 80 percent of rated thermal power). Operations personnel were performing a Turbine Control Valve (TCV EIIS Code JJ) Fast Closure Instrument Functional Test. With the Reactor Protection System (RPS EIIS Code JC) channel A in the tripped condition due to the TCV testing, the RPS channel B unexpectedly tripped, resulting in a scram.

The root cause of this event appears to be equipment failure. Specifically, the probable cause of the RPS channel B trip was a failed 15 volt regulator card in the Average Power Range Monitor (APRM EIIS Code IG) channel B. However, since it was determined that the alarm printer would not have recorded a high pressure scram signal due to a failed process computer board, the RPS channel B trip could also have been caused by a spurious high pressure trip.

Corrective actions for this event included thoroughly investigating trip system B1 (the portion of RPS channel B logic which actually tripped) for the cause of the trip and replacing the failed equipment.

A. REQUIREMENT FOR REPORT

This report is required by 10 CFR 50.73 (a)(2)(iv), because unplanned actuations of the Reactor Protection System (RPS EIIS Code JC) and some Engineered Safety Features (ESFs) occurred. The ESFs that actuated were: 1) Primary Containment Isolation System (PCIS EIIS Code JM) valve Groups 1 and 2, 2) Safety Relief Valve (SRV) Low Low Set (LLS EIIS Code JE), and 3) High Pressure Coolant Injection (HPCI EIIS Code BJ).

B. UNIT(s) STATUS AT TIME OF EVENT

1. Power Level/Operating Mode

Unit 2 was in the run mode at an approximate power level of 1949 MWt (approximately 80 percent of rated thermal power).

2. Inoperable Equipment

There was no inoperable equipment that contributed to this event.

C. DESCRIPTION OF EVENT

1. Event

On 4/17/88 at approximately 0140 CDT, Operations personnel were performing procedure 34SV-C71-005-2S (Turbine Control Valve Fast Closure Instrument Functional Test). This procedure is a normally scheduled surveillance that verifies, among other items: 1) the closure response of the main Turbine Control Valves (TCVs EIIS Code JJ), and 2) that the RPS logic functions as designed when the TCVs fast close (i.e. a half scram is inserted on the appropriate RPS channel).

At 0143 CDT, Operations personnel had satisfactorily completed testing of TCV-1. Personnel began testing of TCV-2. As anticipated, as a result of the testing, the RPS channel A tripped. This half scram signal occurred at approximately 0153 CDT.

With the RPS channel A still in the tripped condition due to the TCV testing, the RPS channel B also tripped. (Subsequently it was determined that trip system B1 was the portion of RPS channel B logic which actually tripped.) Operations personnel did not receive any annunciators or process computer printout that indicated the reason for the RPS channel B trip. The full RPS logic actuation resulted in a reactor scram.

At 0154 CDT, as a result of void collapse due to the scram, the sensed reactor water level decreased from approximately 37 inches above instrument zero to approximately 24 inches below instrument zero. This level (-24 inches) is approximately 140 inches above the Top of Active Fuel (TAF) and was the lowest reactor water level reached during the course of this event.

During the reactor water level decrease, PCIS valve Group 2 isolated, per design, when reactor water level was approximately 12 inches above instrument zero. The Reactor Feed Pumps (RFPs EIIS Code SJ) also sensed the decrease in reactor water level and automatically increased their injection flow rate to quickly restore level. Reactor water level reached 56 inches above instrument zero and the RFPs tripped automatically, per design, on the high reactor water level signal.

The reactor water level continued to swell as expected. However, Operations personnel noted that reactor water level was soon indicating above +60 inches (the maximum level of the normal range reactor water level instruments) and was on scale on the wide range (shutdown flooding) level instrument. This instrument is designed to be used during refueling/shutdown conditions and is not pressure or temperature compensated to provide accurate readings in other operating conditions.

From the information immediately available, Operations personnel were unsure of the exact water level. At 0155 CDT, Operations personnel decided, as a conservative action, to close the Main Steam Isolation Valves (MSIVs EIIS Code JM), in order to prevent any reactor water from entering the Main Steam Lines (MSLs). (Subsequent analysis of the event showed that the maximum water level reached in this event was approximately 70 inches above instrument zero. The bottom of the MSLs is at approximately 100 inches.)

The closure of the MSIVs isolated the reactor pressure vessel from the main condenser (EIIS Code SG). In this configuration scram recovery takes a longer period of time. As reactor pressure increases due to decay heat it can be controlled via use of the LLS system, Reactor Core Isolation Cooling (RCIC EIIS Code BN), and HPCI.

Operations personnel, in accordance with the requirements of the Emergency Operating Procedures (EOPs) in this plant configuration, allowed reactor pressure to increase to fulfill the high pressure portion of the LLS arming logic. At 0208 CDT, reactor pressure reached approximately 1045 psig. A high reactor pressure scram signal was received, per design, and a high pressure signal was input into the LLS arming logic.

Operations personnel next manually opened SRV F, one of the LLS SRVs. This action completed the arming of the LLS logic. SRVs B, G, and D (the other LLS SRVs) then actuated in their LLS mode. As a result of the LLS operation, reactor pressure decreased to approximately 847 psig, and all four of the LLS SRVs reset.

At 0209 CDT, reactor water level had decreased due to the LLS actuation, and a low reactor water level (approximately 12 inches above instrument zero) scram signal was received. (The PCIS valve Group 2 isolation signal, due to low water level, that occurred at 0154 CDT had not been reset. Therefore, the valves were already closed, and no further valve movement resulted.)

Reactor water level continued to decrease to approximately 2 inches above instrument zero. At approximately 0210 CDT, Operations personnel manually initiated the RCIC system to control reactor water level and to remove steam from the reactor vessel. At approximately 0228 CDT, reactor water level reached approximately 51 inches above instrument zero. RCIC automatically tripped, per design, on its high reactor water level signal.

When controlling reactor pressure with LLS, due to the reactor being isolated from the main condenser, swings in reactor pressure (with corresponding changes in reactor level) are expected.

At approximately 0229 CDT, reactor pressure had risen again sufficiently that SRV B opened at its LLS opening pressure setpoint of approximately 997 psig. At approximately 0230 CDT, reactor pressure had decreased such that SRV B closed at its LLS closing pressure setpoint of approximately 851 psig.

At 0232 CDT, a PCIS valve Group 1 logic actuation occurred as a result of a low main condenser vacuum signal. When the MSIVs were closed (at approximately 0155 CDT), the Steam Jet Air Ejectors (SJAE EIIS Code SH) lost their source of steam. (The SJAEs are used to maintain main condenser vacuum during normal operations. They use nuclear steam to accomplish this function and the steam supply for the SJAEs is tied into a MSL downstream of the outboard MSIV.) As a result, the vacuum in the main condenser decreased to approximately 10 inches Hg vacuum, and the PCIS valve Group 1 isolation occurred, per design.

At approximately 0233 CDT, the reactor water level had again decreased to approximately 20 inches above instrument zero due to the functioning of LLS at 0229 CDT. However, by 0245 CDT, level had been restored to 49 inches above instrument zero by water from the Control Rod Drive (CRD EIIS Code AA) system.

Also, at approximately 0245 CDT, reactor pressure had increased sufficiently again for SRV B to actuate in the LLS mode. At approximately 0246 CDT, after relieving reactor pressure, SRV B closed again. Due to this SRV actuation, reactor water level decreased to 20 inches above instrument zero by 0248 CDT, and again it was restored to approximately 32 inches, at 0255 CDT, via the CRD system.

At approximately 0255 CDT, SRV B again opened to relieve pressure in its LLS mode, and it closed at 0256 CDT. Due to this SRV actuation water level again decreased. At approximately 0256 CDT, reactor water level reached approximately 12 inches above instrument zero. Both the RPS and PCIS valve Group 2 again received actuation signals, per design, on low reactor water level. At 0258 CDT, reactor water level reached its lowest level during this pressure reduction evolution of approximately 5 inches above instrument zero.

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

At 0305 CDT, Operations personnel were in the final phases of stabilizing the plant. This was accomplished by manually injecting water into the reactor using the RCIC system (at 0305 CDT), and using RCIC to remove excess steam. To further aid in the removal of steam, Operations personnel manually initiated the HPCI system in the full flow test mode (at 0305 CDT). (In this mode, HPCI takes a suction from the Condensate Storage Tank [CST EIIS Code KA] and discharges back to the CST. Steam is extracted from the reactor vessel to run the HPCI pump.)

Also at 0305 CDT, reactor water level had increased to approximately 11 inches above instrument zero.

By approximately 0320 CDT, reactor water level had stabilized in the normal operating range (at approximately 40 inches above instrument zero). Operations personnel continued to control vessel cooldown with RCIC and HPCI. No more actuations of LLS valves were needed.

At 0422 CDT, the NRC was notified of the actuation of the RPS and the resulting ESF actuations, per 10 CFR 50.72 reporting requirements. An investigation to determine the cause of the scram was initiated, per the plant's administrative control requirements.

2. Dates/Times

| Date | Time (CDT) | Description |
|---|---|---|
| 4/17/88 | 0140 | Operations personnel were performing procedure 34SV-C71-005-2S (Turbine Control Valve Fast Closure Instrument Functional Test). |
| | 0143 | Fast closure test of TCV-1 was satisfactorily completed. Operations personnel began to test TCV-2. |
| | 0153 | Due to testing TCV-2, RPS channel A tripped. |
| | | Received a half scram signal in RPS channel B, with the cause unknown and while the scram signal in RPS channel A was still present. A reactor scram occurred. |

| Date | Time (CDT) | Description |
|---|---|---|
| 4/17/88 | 0154 | Water level dropped to 24 inches below instrument zero due to void collapse. During the level decrease, a PCIS valve Group 2 isolation was received at approximately 12 inches above instrument zero. |
| | | The RFPs quickly restored water level and tripped on high reactor water level (56 inches above instrument zero). |
| | 0155 | Indicated reactor water level was now greater than 60 inches above instrument zero. Operations personnel were unsure of the exact water level and closed the MSIVs to prevent water entry into the MSLs. |
| | 0208 | In accordance with the EOPs, Operations personnel allowed reactor pressure to reach approximately 1045 psig. A high pressure scram signal was received and the reactor pressure portion of the LLS arming logic was fulfilled. |
| | | Operations personnel manually opened SRV F in order to complete the arming of the LLS logic. SRVs B, G, and D then operated in the LLS mode to relieve pressure and closed at approximately 847 psig. |
| | 0209 | Reactor water level decreased, due to the LLS actuation, to approximately 2 inches above instrument zero. At approximately 12 inches above instrument zero, RPS and PCIS valve Group 2 received actuation signals (Group 2 valves already closed due to previous isolation signal). |

| Date | Time (CDT) | Description |
|---|---|---|
| 4/17/88 | 0210 | Operations personnel manually initiated RCIC to restore level. |
| | 0228 | RCIC trips on a high water level of 51 inches above instrument zero, per design. |
| | 0229 | SRV B again operated in its LLS mode to relieve reactor pressure. |
| | 0230 | SRV B closed at its LLS closing pressure setpoint. |
| | 0232 | A PCIS valve Group 1 isolation signal was received on low condenser vacuum due to loss of SJAEs, caused by previous closing of MSIVs. |
| | 0233 | Again, due to the LLS actuation (at 0229 CDT), reactor water level had decreased to 20 inches above instrument zero. |
| | 0245 | Water level was restored to 49 inches above instrument zero due to CRD system flow. |
| | | SRV B again operated in its LLS mode to relieve reactor pressure. |
| | 0246 | SRV B closed at its LLS closing pressure setpoint. |
| | 0248 | Water level had dropped to 20 inches above instrument zero. |
| | 0255 | Water level was restored to 32 inches above instrument zero due to CRD system flow. |
| | | SRV B again operated in its LLS mode to relieve reactor pressure. |

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

| Date | Time (CDT) | Description |
|---|---|---|
| 4/17/88 | 025? | SRV B closed at its LLS closing pressure setpoint. |
| | | Level decreased to approximately 12 inches above instrument zero; RPS and PCIS valve Group 2 again received actuation signals. |
| | 0258 | Reactor water level reached 5 inches above instrument zero. |
| | 0305 | Operations personnel again started RCIC injection to control level. Operations personnel also started HPCI in full flow test mode to control reactor pressure. Water level had increased to 11 inches above instrument zero. |
| | 0320 | Reactor water level had stabilized in the normal operating range (at approximately 40 inches above instrument zero). Operations personnel continued to control vessel cooldown with RCIC and HPCI. No more actuations of LLS valves were needed. |
| | 0422 | The NRC was notified of the actuation of the RPS and the resulting ESF actuations, per 10 CFR 50.72 reporting requirements. |
| | | An investigation to determine the cause of the scram was initiated. |

3. Other Systems Affected

No systems, other than the RPS, PCIS valve Groups 1 and 2, LLS, RCIC, and HPCI, were affected by this event. These systems have no secondary functions.

4. Method of Discovery

Licensed Operations personnel discovered the full logic actuation of the RPS and resulting scram by observation of control room indications.

5. Operator Actions

Operations personnel performed the following actions:

1. Conservatively closed the MSIVs to ensure that no water entered the MSLs.

2. Responded to plant conditions in accordance with the EOPs.

6. Auto/Manual Safety System Response

The RPS, PCIS, and LLS system actuated automatically. Plant operations personnel manually initiated the HPCI system (in the full flow test mode) and opened one of the LLS SRVs to arm the LLS system.

D. CAUSE OF EVENT

1. Immediate Cause

The immediate cause of this event was an unanticipated trip of the RPS channel B (trip system B1 was the portion of RPS channel B logic which actually tripped), while RPS channel A was tripped (as a result of TCV testing). The TCV testing was being performed per plant procedures. With both RPS channels A and B tripped, a full reactor scram occurred.

2. Root/Intermediate Cause

The unanticipated trip of RPS trip system B1 was thoroughly investigated with the following results:

1. No procedural problems were found with 34SV-C71-005-2S (Turbine Control Valve Fast Closure Instrument Functional Test). This procedure was reperformed with acceptable results while Unit 2 was shutdown.

2. Functionally tested all inputs to RPS trip system B1. Verified all inputs properly caused a B1 trip, annunciated, and printed on the alarm typer. All inputs worked properly except the reactor high pressure trip which would not print on the alarm printer (the trip and annunciation worked correctly). The cause of this was determined to be a failed process computer board. The process computer board was replaced.

3. Visually inspected RPS relay panels 2H11-P608 and 2H11-P611 (which contain RPS trip system B1 logic) for loose wires, open links or other deficiencies with no problems identified.

4. Performed voltage checks in RPS trip system B1 with all results acceptable.

5. General Electric Company (GE) was contacted to assist in the investigation of the trip of RPS trip system B1. GE determined that a 15 volt regulator card in the Average Power Range Monitor (APRM EIIS Code IG) channel B had a mechanical sensitivity which caused loss of one of its regulated outputs. Loss of this output could result in the trip effects seen. The APRM channel B provides input to RPS trip system B1.

Based on the results of the investigation, two possible root causes for the RPS trip system B1 trip were identified. The first possible root cause is a spurious high reactor pressure trip. At the time the RPS trip system B1 trip was received, the event investigation verified that reactor pressure was in the normal operational range, well below the reactor high pressure scram setpoint of 1045 psig. However, the receipt of a spurious high pressure trip cannot be ruled out, since the alarm printer would not have recorded such a trip due to a failed process computer board (failure attributed to normal end of life).

The more probable root cause of the event is the failed 15 volt regulator card (failure attributed to normal end of life) in the APRM channel B circuit. The failure mode of the regulator card could have caused the APRM channel B to trip and then clear before the process computer could identify the cause of the trip. The RPS would have responded to the brief false high flux signal; however, the trip would not have printed on the alarm printer.

E. ANALYSIS OF EVENT

The RPS provides timely protection against the onset and consequences of conditions that could threaten the integrities of the fuel barriers and the nuclear system process barrier.

In this event, the initial RPS actuation occurred when RPS channel B tripped unexpectedly (due either to a spurious high pressure trip or a failed component in the APRM system) while RPS channel A was in a tripped condition due to testing of the TCVs. Although the input signals to the RPS were not reflective of actual plant conditions, the RPS functioned conservatively, as designed.

During the recovery from the scram, reactor water level appeared to rise higher than would be normally expected. Operations personnel took the conservative action of closing the MSIVs to prevent possible entry of reactor water into the MSLs. Subsequent analysis demonstrated that the maximum water level reached in this event was approximately 70 inches above instrument zero. The bottom of the MSLs is at approximately 100 inches.

However, the closure of the MSIVs isolated the reactor pressure vessel from the main condenser. In this configuration scram recovery takes a longer period of time. As reactor pressure increases due to decay heat the LLS, RCIC, and HPCI systems are available to control reactor pressure and water level.

The LLS relief logic system is designed to mitigate the thrust loads on the Safety Relief Valve Discharge Lines (SRVDLs) and the resulting loads on the torus shell from subsequent SRV actuations. The LLS system ensures that subsequent SRV actuations occur after the water leg in the SRVDL stabilizes, at its normal level, by increasing the blowdown range and decreasing the closing and opening setpoints for the four LLS SRVs.

In this event LLS functioned, as designed, to control reactor pressure upon manual arming of the system by Operations personnel in accordance with the EOPs.

The HPCI system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperatures in the event that reactor water level decreases without a rapid depressurization of the reactor vessel. The HPCI system is capable of maintaining sufficient reactor vessel inventory until the vessel is depressurized.

In this event, flow from the CRD and RCIC systems were capable of restoring reactor water level in a timely manner. However, HPCI was used in its full flow test mode to help control reactor pressure. In this mode the HPCI turbine (which drives the HPCI pump) draws steam from the reactor vessel and the pump takes suction from the CST and discharges back to the CST.

Finally, during the scram recovery the PCIS valve Groups 1 and 2 responded, as designed, to respective signals of low condenser vacuum and low reactor water level. The low condenser vacuum resulted when the SJAEs lost their source of steam. They draw steam from an MSL downstream of the outboard MSIV and thus could not continue to perform their function of maintaining condenser vacuum when the MSIVs were closed. The low reactor water level occurred during the swings in level which are expected when LLS actuates.

The PCIS provides timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and the nuclear system process barrier (such as the reactor vessel). This protection is accomplished by the isolation of process lines that penetrate the primary containment. In this event, the signals to which PCIS responded were not due to a breach in the nuclear system process barrier; however, PCIS responded conservatively.

Based on the above information, it is concluded that this event had no adverse impact on nuclear plant safety. Additionally, the above analysis is applicable to all power levels.

F. CORRECTIVE ACTIONS

The corrective actions for this event included:

1. Functionally testing all inputs to RPS channel B1.

2. Visually inspecting RPS relay panels 2H11-P608 and 2H11-P611 for deficiencies.

3. Performing voltage checks in RPS channel B1.

4. Replacing the failed 15 volt regulator card in the APRM channel B circuit per Maintenance Work Order 2-88-2141.

5. Replacing the failed process computer board which would have prevented the alarm printer from recording a high reactor pressure scram signal.

G. ADDITIONAL INFORMATION

1. FAILED COMPONENT(s) IDENTIFICATION

Regulator Card
MPL (Plant Index Identifier): 2C51-AR32 VR-3
Manufacturer: General Electric Company
Model Number: 136B2772AAG001
Type: RG
EIIS: IG

Process Computer Board
MPL (Plant Index Identifier): 2C92
Manufacturer: General Electric Company
Model Number: 117C3328G001
Type: Input Signal Board
EIIS: ID

2. PREVIOUS SIMILAR EVENTS

There has been one similar event to the one described in this LER. It was reported in LER 50-366/1987-003 (dated 1/26/87).

This LER describes an event where a test procedure step was in progress that, in conjunction with an unexpected equipment failure, caused closure of the MSIVs and a reactor scram since the reactor was in the run mode.

The root cause of the event was equipment failure (a temperature switch monitor). The corrective actions included replacement of the temperature sensor and revision of the test procedure involved.

However, the corrective actions for the similar event would not have prevented the event described by LER 50-366/1988-011 because the cause of the similar event was different as noted above.