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) |
|---|---|---|
| PLANT VOGTLE - UNIT 1 | 0 5 0 0 0 4 2 4 | 1 OF 0 3 |

TITLE (4)

PRESSURE TRANSMITTER FAILURE CAUSES ESF ACTUATION ON STEAM GENERATOR HI-HI WATER LEVEL

| 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 5 0 0 0 |
| 0 6 | 2 0 | 8 7 | 8 7 | — | 0 3 9 | — | 0 0 | 0 7 | 2 0 | 8 7 | | 0 5 0 0 0 |

| OPERATING MODE (9) | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11) | | | |
|---|---|---|---|---|
| 3 | 20.402(b) | 20.405(c) | X 50.73(a)(2)(iv) | 73.71(b) |
| POWER LEVEL (10) 0 0 0 | 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 |
| W. E. Burns, Nuclear Licensing Manager - Vogtle | 4 0 4 5 2 6 - 7 0 1 1 4 |

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 | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| B | S J B | I P T | R 3 6 9 | Y | | | | | | | |
| | | | | | | | | | | | |

| 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 June 20, 1987, at 1552 CDT with Unit 1 in Mode 3, a main feedwater (MFW) isolation, trip of both MFW pumps and subsequent motor-driven auxiliary feedwater (AFW) pump start signal occurred due to steam generator (SG) #2 reaching its High-High water level setpoint. The SG High-High level was caused by an atmospheric relief valve (ARV) that had lifted and remained open. The resultant pressure drop caused the SG water volume to swell. The motor driven AFW pumps were being used for SG level control; the "A" MFW pump was operating in the recirculation mode and the "B" MFW pump was in the tripped condition. Plant operators manually closed the ARV and manually controlled the motor driven AFW pumps to stabilize plant conditions.

A failed pressure transmitter (PT) caused the opening of the ARV. The failed pressure transmitter was replaced.




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| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) | | | PAGE (3) |
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| PLANT VOGTLE - UNIT 1 | 0 5 0 0 0 4 2 4 | 8 7 | — 0 3 9 | — 0 0 | 0 2 OF 0 3 |

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

A. REQUIREMENT FOR REPORT

This report is required per 10 CFR 50.73 (a)(2)(iv) because the initiation of the motor driven auxiliary feedwater pumps constituted an unplanned Engineered Safety Feature (ESF) system actuation .

B. UNIT STATUS AT TIME OF EVENT

Unit 1 was in Mode 3 (Hot Standby) at 0% of rated thermal power (RTP). Reactor coolant system (RCS) pressure and temperature were approximately 2235 psig and 557 degrees Fahrenheit, respectively.

C. DESCRIPTION OF EVENT

On June 20, 1987, the plant was in Mode 3 while an investigation was being conducted to determine the cause of a prior Reactor Protection System (RPS) actuation (Reference: LER 50-424/1987-038). The steam generators (SGs) were being fed by the motor driven auxiliary feedwater (AFW) pumps. The "A" main feedwater (MFW) pump was operating in the recirculation mode for testing and the "B" MFW pump was in the tripped condition. At 1552 CDT, a false high pressure signal from a pressure transmitter (1-PT-3010) in the steam header caused the loop 2 atmospheric relief valve (ARV) to lift and remain in its full open position, dropping steam header pressure and causing the SG swell, which resulted in the SG #2 High-High water level signal. SG #2 exceeded its High-High level setpoint causing a MFW isolation signal, turbine trip signal, a trip signal for both MFW pumps and a subsequent motor-driven AFW pump actuation signal. As previously discussed, the motor driven AFW pumps were being used to supply water to the steam generators. The "A" MFW pump tripped and all valves isolated as designed. Operators checked other pressure indicators in the steam header, then manually shut the ARV and throttled the AFW flow control valves. A work order was written to determine the cause of the false high pressure indication.

D. CAUSE OF EVENT

The High-High water level in SG#2 was caused by a swelling of SG water level. A pressure drop in the SG was caused by an ARV lifting and remaining fully open on receipt of a false high pressure indication generated from a faulty pressure transmitter (PT). An investigation found an output current of 23 milli-amperes for this PT (1PT-3010). The normal output range of this type PT is 4-20 milli-amperes. The PT was replaced and the orginal PT was

bench tested for an 11-day period, however, the output current remained in the normal range. It is believed that a component of one of two printed circuit boards inside of the PT fails intermittently. Since a reason for the intermittent failure cannot be found, the failure mode is regarded as spurious.

E. ANALYSIS OF EVENT

When the SG #2 water level rose to its High-High water level setpoint, the MFW isolation and AFW actuation signals were initiated and the respective equipment functioned as designed to protect the plant. Since these ESF's functioned as designed, a similar event at a higher power level would be expected to have a similar result, ensuring both plant safety and public health and safety.

F. CORRECTIVE ACTIONS

The pressure transmitter was replaced and bench tested with no conclusive results. This is the only failure of this model pressure transmitter.

G. ADDITIONAL INFORMATION

1. Failed Component
   Pressure Transmitter Manufactured by Rosemount Engineering
   Model #1153-GB9

2. Previous Similar Events
   None

3. Energy Industry Identification System Codes
   Auxiliary Feedwater System - BA
   Condensate and Feedwater System - SJ
   Main Steam System - SB

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the southern electric system

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July 20, 1987

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

PLANT VOGTLE - UNIT 1
NRC DOCKET 50-424
OPERATING LICENSE NPF-68
LICENSEE EVENT REPORT
PRESSURE TRANSMITTER FAILURE CAUSES ESF
ACTUATION ON STEAM GENERATOR HIGH-HIGH WATER LEVEL

Gentlemen:

Pursuant to the requirements of 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting a Licensee Event Report (LER) concerning an event where an ESF actuation was caused by the failure of a pressure transmitter.

Sincerely,

William F. Bruen /for

L. T. Gucwa

PAH/lm

Enclosure: LER 50-424/1987-039

c: (see next page)

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Georgia Power



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