PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Salem Generating Station

October 04, 1990

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

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-75
DOCKET NO. 50-311
UNIT NO. 2
LICENSEE EVENT REPORT 90-036-00

This Licensee Event Report is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR 50.73(a)(2)(iv). This report is required within thirty (30) days of discovery.

Sincerely yours,

S. LaBruna
General Manager -
Salem Operations

MJP:pc

Distribution



IE22
1/1

NRC Form 366 (9-83)

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

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Salem Generating Station - Unit 2 | 0 5 0 0 0 3 1 1 | 1 OF 0 5 |

TITLE (4)
Turbine Trip/Rx Trip on No. 24 S/G Hi-Hi Level (P-9) Due To Equipment Failure

| 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 9 | 0 4 | 9 0 | 9 0 | 0 3 6 | 0 0 | 1 0 | 0 4 | 9 0 | | 0 5 0 0 0 |

| OPERATING MODE (9) | 1 | THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| POWER LEVEL (10) | 0 6 0 | 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 |
| M. J. Pollack - LER Coordinator | 6 0 9 3 3 9 - 2 0 2 2 |

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 | 6 3 | U 0 8 0 | Y | | B | S J | F C O | H 0 1 5 | Y | |
| B | S D | V | M 1 2 0 | 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 9/4/90 at 0226 hours, during normal power operation, No. 21 Steam Generator Feed Pump (SGFP) tripped on Low Suction Pressure. A turbine runback to 60% power at 15%/min was initiated. At the completion of the runback, Steam Generator (S/G) level began increasing. As per procedure, the four BF19 valves (S/G Feedwater Control Valves) were placed in manual mode to gain control of S/G level. However, just after the 24BF19 valve was put in manual control, No. 24 S/G reached its high level setpoint initiating a Turbine Trip at 0231 hours on 9/4/90. With reactor power above permissive P-9 (50% power), a reactor trip followed. The root cause of the reactor trip is attributed to equipment failure. The 21BF19 Auto/Man controller failed, causing valve closure during the recovery operation after the No. 21 SGFP had tripped. The setpoint for the suction pressure switch had drifted high. When the 23HD15 valve (Heater Drain Pump Discharge Control Valve) failed closed (ruptured diaphragm) a significant SGFP suction pressure drop occurred. This coupled with the high setpoint resulted in the No. 21 SGFP trip. Preceding the rupture of the valve diaphragm, the valve controller failed. The failed pressure switch, the 23HD15 valve diaphragm, and the 23HD15 valve controller's pneumatic relay were replaced. The 21BF19 valve Auto/Man controller was replaced. The preventive maintenance program for the HD15 valves will be reviewed. The secondary plant process controls will be evaluated to ensure optimum reliability with minimal operator required intervention. A review of existing procedural guidance for actions following a loss of a SGFP during full power operation has been initiated.

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as {xx}

IDENTIFICATION OF OCCURRENCE:

Turbine Trip/Reactor Trip from 60% power on No. 24 Steam Generator Hi-Hi Level due to equipment failure

Event Date: 9/04/90

Report Date: 10/04/90

This report was initiated by Incident Report No. 90-652.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 100% - Unit Load 1100 MWe
Makeup being provided to the No. 22 Condenser Hotwell; No. 21B Circulator out of service

DESCRIPTION OF OCCURRENCE:

On September 4, 1990 at 0226 hours, during normal power operation, No. 21 Steam Generator Feed Pump (SGFP) {SJ} tripped on Low Suction Pressure. In accordance with procedure, Operations initiated a turbine runback to 60% power at 15%/min. At the completion of the runback, Steam Generator (S/G) level indicated approximately 24%. With the No. 22 SGFP running at full speed, S/G level began increasing. As per Abnormal Operating Procedure AOP-CN-1, the four (4) BF19 valves (S/G Feedwater Control Valves) were being placed in manual mode to gain control of S/G level. However, just after the 24BF19 valve was put in manual control, No. 24 S/G reached its high level setpoint initiating a Turbine Trip at 0231 hours on September 4, 1990. With reactor power above permissive P-9 (50% power), a reactor trip followed.

The Unit was stabilized in Mode 3 (Hot Standby). Subsequently, on September 4, 1990 at 0308 hours, the Nuclear Regulatory Commission (NRC) was notified of the automatic actuation of the Reactor Protection System {JC} in accordance with the Code of Federal Regulations 10CFR 50.72(B)(2)(ii).

APPARENT CAUSE OF OCCURRENCE:

The root cause of the reactor trip is attributed to equipment failure.

This event initiated with the trip of the No. 21 SGFP. Following the loss of the No. 21 SGFP, a turbine runback was successfully completed to 60% power. However, S/G level increased to the No. 24 S/G Hi-Hi Level turbine trip setpoint. Since power was above 50% power (P-9),

APPARENT CAUSE OF OCCURRENCE: (cont'd)

a reactor trip followed. Investigation revealed that when the Operator placed the 21BF19 valve control in manual (first valve to be so placed) the valve closed. A "bumpless" transfer from automatic to manual control did not occur. Bumpless transfer is designed to provide a smooth transition from automatic to manual control without causing a change in valve position. It did not occur do to a malfunction of the 21BF19 Man/Auto controller which was due to a failure of its internal electronic components.

Without the bumpless transfer the 21BF19 valve closed causing an increase of feedwater flow to the other S/Gs. The 21BF19 valve control was regained by the Operator. Nos. 22BF19 and 23BF19 were able to be placed in manual control; however, No. 24 S/G reached its Hi-Hi Level setpoint just after the 24BF19 valve was placed in manual and S/G level control had been initiated.

The cause of the No. 21 SGFP trip is attributed to a combination of equipment failures. The 2PD-581 SGFP suction pressure switch setpoint was found high. This coupled with the pressure transient caused by the closure of the 23HD15 valve (Heater Drain Pump Discharge Control Valve) caused the trip of the SGFP.

The normal setpoint for the suction pressure switch SGFP Low Suction Pressure trip is 215 psig. The setpoint was found to be 309 psig. Investigation revealed that the switch had failed resulting in significant setpoint drift. This drift was able to be repeated by testing. It would drift high approximately 60 to 80 psig.

Normal SGFP suction pressure Control Room indication is between 360 and 380 psig with the plant at 100% power. Actual SGFP suction pressure is approximately 20 psig less. This Control Room indication's P-250 computer sensing point is located on the common SGFP line prior to the line splitting to the individual SGFPs. The suction pressure alarm sensing switch is located on each independent SGFP suction line. This causes the alarm sensing switch to see lower pressure due to various pressure reducing factors (i.e., strainer, valve, orifice and pipe elbows.

The cause of the suction pressure drop to the SGFP was the closure of the 23HD15 valve (Heater Drain Pump Discharge Control Valve). Investigation identified that the valve diaphragm had ruptured. Also, the reset portion of the valve controller (2LA1022C) was found failed. Based upon the post trip review, it is concluded the controller failed approximately two (2) hours prior to the SGFP trip when the condensate polisher flows had begun to oscillate. The valve eventually closed when the diaphragm ruptured. The closure of the valve resulted in a pressure transient similar to the loss of a Heater Drain Pump which, based upon past experience, causes a 30 to 40 psig drop in SGFP suction pressure. Therefore, with the suction pressure trip setpoint high and the suction pressure at approximately 340 psig, the pressure transient, caused by the closure of the 23HD15 valve, was sufficient to trip the No. 21 SGFP.

APPARENT CAUSE OF OCCURRENCE: (cont'd)

The function of the HD15 valve is to maintain level in the Moisture Separator Reheater and Feedwater Heater Drain Tank. The tank collects condensate from various sources and directs its discharge, via the Heater Drain Pump, to the SGFP suction header.

ANALYSIS OF OCCURRENCE:

The turbine trip on Steam Generator Hi-Hi level is an anticipatory trip to prevent moisture carryover from the Steam Generator and damage to the main turbine from excessive moisture carryover. The purpose of the reactor trip is to reduce the primary plant transient resulting from the loss of the turbine above 50% power. A turbine trip above 50% power results in a direct reactor trip and a controlled short term release of steam to the condenser via the steam dumps. This steam release removes sensible heat from the reactor coolant system and precludes Steam Generator safety valve operation.

The reactor trip after a turbine trip with the Unit above 50% power is anticipatory and is included as part of good engineering practice and prudent design. No credit is taken in any of the safety analysis for this trip. For rapid transients during power operation, protection of the reactor is provided by the power range detectors. For slower developing transients, protection of the reactor is provided by overtemperature and overpower delta temperature bistables.

The Reactor Protection System functioned as designed. A Main Feedwater Isolation (an Engineered Safety Feature) followed the trip as per design. Therefore, no undue risk to the health and safety of the public was involved. However, because of the automatic actuation of the Reactor Protection System, the event is reportable in accordance with the Code of Federal Regulations 10CFR 50.73 (a)(2)(iv).

A review of operator performance, relative to the circumstances associated with this event, was conducted. It was found to be satisfactory.

CORRECTIVE ACTION:

The SGFP pressure switch has been replaced with an equivalent switch. The particular model switch is no longer manufactured. The preventive maintenance program for the equivalent switch will be reviewed to ensure reliable operation as determined from its mean time between failures.

An historical review of the failed SGFP pressure switch was conducted. The historical switch calibration data did not show a significant drift concern. Also, an investigation of the failed SGFP pressure switch was conducted using the INPO CFAR System (Component Failure Analysis Reporting System). The failure rate for Salem Station is within the industry standard. This switch failure has been evaluated to be a random equipment failure.

CORRECTIVE ACTION: (cont'd)

Investigation of the problem with switching the 21BF19 valve control from automatic to manual was conducted. The 21BF19 Man/Auto controller was found to be malfunctioning, preventing a bumpless transfer. This was due to a failure of its internal electronic components. It was subsequently replaced.

The 23HD15 valve actuator diaphragm was replaced. The valve was subsequently successfully stroked and declared operable.

The preventive maintenance program for the HD15 valves will be reviewed. Modifications to the program will be made as applicable.

The 23HD15 valve controller's pneumatic relay was found failed. It was replaced.

The secondary plant process controls will be evaluated to ensure optimum reliability with minimal operator required intervention.

Operations Department management has initiated a review of the existing procedural guidance for actions following a loss of a SGFP during full power operation. Each SGFP is capable of supplying 65% of full power feedflow requirements.

Failure Data:

Pressure Switch

Manufacturer: UE
Model: J11

21BF19 man/auto controller

Manufacturer: Hagan
Model: 41180-001

23HD15 Valve

Manufacturer: Masoneillan Int.
Model: 37-10134

V. J. Paluzzi for
General Manager -
Salem Operations

MJP:pc

SORC Mtg. 90-138