NON-PUBLIC?: N
ACCESSION #: 8905170078
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Salem Generating Station - Unit 2 PAGE: 1 OF 4

DOCKET NUMBER: 05000311

TITLE: Rx. Trip from 100% Power due To An Equipment Failure
EVENT DATE: 04/11/89 LER #: 89-008-00 REPORT DATE: 05/10/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M. J. Pollack - LER Coordinator TELEPHONE: (609) 339-4022

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: JG COMPONENT: 74 MANUFACTURER: S440
B IG RJX P323
B SB RV C170
REPORTABLE TO NPRDS: Y
Y
Y

SUPPLEMENTAL REPORT EXPECTED: NO EXPECTED SUBMISSION DATE:

ABSTRACT:

On April 11, 1989 at 2043 hours, a reactor trip on No. 24 Steam Generator (S/G) "Low-Low Level" occurred. The No. 24 main steamline isolation valve, 24MS167, had closed. At the time of the event, the required Technical Specification Surveillance 4.0.5-V for the 21MS18 Main Steamline Bypass Stop Valve was in progress. The root cause of this event has been attributed to an equipment failure. An "operate reset" latching relay (74-4A), associated with the Solid State Protection System (SSPS), failed during the 21MS18 surveillance resulting in closure of the 24MS167 valve. The failed "74-4A" relay was replaced. The circuit was then tested satisfactorily. An inspection of the other 74-4A and 3-3B MS167 relays will be conducted to determine if similar mechanical defects are present. Following the reactor trip, the 24MS15, Main Steamline Safety Valve, lifted twice. The valve remained partially open for approximately one hour. Investigation revealed that the as found lift set pressure was 901 psig (after the second lift). A

proper lift setpoint of 1070 psig could not be achieved. The valve was subsequently replaced with a spare valve. It appears that the valve seating surface during the first lift became damaged allowing increased leakage. The replaced 24MS15 valve has been shipped to Crosby Valve & Gage Co. for inspection and repair.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

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

IDENTIFICATION OF OCCURRENCE:

Reactor Trip From 100% Power Due To An Equipment Failure

Event Date: 4/11/89

Report Date: 5/10/89

This report was initiated by Incident Report No. 89-227.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 100% - Unit Load 1140 MWe

DESCRIPTION OF OCCURRENCE:

On April 11, 1989 at 2043 hours, a reactor trip on No. 24 Steam Generator (S/G) "Low-Low Level" occurred. The No. 24 main steamline isolation valve, 24MS167, SB had closed. At the time of the event, the required Technical Specification Surveillance 4.0.5-V for the 21MS18 Main Steamline Bypass Stop Valve was in progress.

The Unit was stabilized in Mode 3 (Hot Standby) and at 2152 hours, on April 11, the Nuclear Regulatory Commission was notified of the actuation of the Reactor Protection System JC in accordance with Code of Federal Regulations 10CFR 50.72 (b) (2) (ii).

APPARENT CAUSE OF OCCURRENCE:

The root cause of this event has been attributed to an equipment failure.

As stated above, surveillance testing of valve 21MS18 was in progress prior to the event. When the Solid State Protection System (SSPS) JG Train "A" output interface cabinet switch was turned to "operate output", the 24MS167 valve closed followed by the trip. The "operate output" function causes closure of the MS18 valve while checking continuity of the MS167 valve closure circuit. Investigation revealed that the 74-4A relay (an operate reset latching relay) was resetting too quickly. The coil mounting bracket within the relay was found deformed allowing the contacts to be made up very fast. When the interface cabinet switch was turned, the relay actuated before the voltage across the relay reset coil had time to drop below the pickup value.

ANALYSIS OF OCCURRENCE:

The Low-low S/G level reactor trip prevents operation with the S/G

ANALYSIS OF OCCURRENCE: (cont'd)

water level below the minimum volume required for adequate heat removal; thereby preventing the loss of the reactor heat sink. The trip is actuated on two out of three low-low water level signals in any S/G. The setpoint ensures that there is adequate inventory in the S/Gs, at the time of the reactor trip, to allow for any possible starting delays of the Auxiliary Feedwater (AFW) System BA ; thus preventing S/G dry-out and the Reactor Coolant System (RCS) AB thermal and hydraulic transients that would be associated with a loss of the heat sink following quick closure of the MS167 valve.

Following the reactor trip, the 24MS15, Main Steamline Safety Valve, lifted twice. The first valve lift occurred at its setpoint of 1070 psig after which it seated properly. Approximately one hour after the first lift, the second lift occurred at a steamline pressure of 1010 psig. The valve remained partially open for approximately one hour. The Reactor Coolant System (RCS) AB average temperature was reduced to 535 deg. F to have the valve fully reseat. Investigation revealed that the as found lift set pressure was 901 psig. A proper lift setpoint of 1070 psig could not be achieved. The valve was subsequently replaced with a spare valve. It appears that the valve seating surface during the first lift became damaged and the valve leakage increased. The design of the valve is such that increased leakage will conservatively cause a decrease in the lift setpoint.

By design, the two Nuclear Instrumentation System (NIS) IG Source Range Detectors (2N31 and 2N32) actuate when reactor power decreases below 6x10(-11) Amps (Permissive P-6), as detected by the Intermediate Range Detectors. However, Source Range Detector 2N32 failed to energize after the reactor trip occurred. Detector 2N31 functioned as designed. The 2N32 NI drawer power supply "crowbar" circuit de-energized the detector and its related equipment. The crowbar circuit is designed to protect the system circuits from excessive power surges.

As per the Technical Specifications, in Modes 3, 4 or 5 only one source range detector is required. Therefore, the inoperability of the 2N32 channel did not affect any safety considerations as analyzed in the Updated Final Safety Analysis Report (UFSAR).

Other than as noted above, the Reactor Protection System (RPS) JC functioned as designed, and the heat sink was maintained. The RCS has been designed to withstand the thermal and hydraulic effects of four hundred (400) trips from full power. This trip was well within the design limits of the system. This occurrence involved no undue risk to the health or safety of the public; however, due to the automatic actuation of the RPS, this event is reportable in accordance with Code of Federal Regulations 10CFR 50.73(a)(2)(iv).

CORRECTIVE ACTION:

The failed 74-4A relay was replaced. The circuit was then tested satisfactorily.

CORRECTIVE ACTION: (cont'd)

An inspection of the other 74-4A and 3-3B MS167 relays will be conducted to determine if similar mechanical defects are present. The Unit 1 relays will be inspected during the current refueling outage and the Unit 2 relays will be inspected during the next outage of sufficient duration.

The replaced 24MS15 valve has been shipped to Crosby Valve & Gage Co. for inspection and repair.

The 2N32 NI drawer power supply was replaced. The circuitry was functionally tested satisfactorily and subsequently declared operable.

FAILURE DATA:

74-4A Relay 24MS15 Safety Valve

Struthers Dunn Crosby Valve & Gage Co.
Relay P/N A255XCX-P Style HA65W, Size 6R10

2N32 HV Power Supply

P/N UPMD-X54
Power Design Inc.

General Manager -
Salem Operations

MJP:pc

SORC Mtg. 89-041

ATTACHMENT 1 TO 8905170078 PAGE 1 OF 1

PSE&G
Public Service Electric and Gas Company
P.O. Box E Hancocks Bridge, New Jersey 08038

Salem Generating Station

May 10, 1989

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 89-008-00

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 being submitted within thirty (30) days of discovery.

Sincerely yours,

L. K. Miller
General Manager -

Salem Operations

MJP:pc

Distribution

The Energy People

*** END OF DOCUMENT ***