NON-PUBLIC?: N
ACCESSION #: 9206010033
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Salem Generating Station - Unit 2 PAGE: 1 OF 07

DOCKET NUMBER: 05000311

TITLE: Reactor Trip, during startup, on No. 24 Steam Generator Low-Low Level.
EVENT DATE: 04/26/92 LER #: 92-007-00 REPORT DATE: 05/22/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 2 POWER LEVEL: 004

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) 339-2022

COMPONENT FAILURE DESCRIPTION:
CAUSE: E,B SYSTEM: SJ COMPONENT: V MANUFACTURER: C635
B SJ V L200
REPORTABLE NPRDS: Y
Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 4/26/92, at 0219 hours, during plant startup, a Reactor Trip occurred on No. 24 Steam Generator (S/G) Low-Low Level during startup following the sixth refueling outage. Prior to the trip, at 0153 hours, with reactor power at 3%, a Turbine Trip signal and Main Feedwater isolation on No. 23 S/G high-high level occurred. Assessment of the cause of the High-High Level turbine trip signal, by the Senior Nuclear Shift Supervisor (SNSS), was that the Nuclear Control Operator (NCO) overfed No. 23 S/G. Approximately 12 minutes later the startup resumed. Reactor power remained at approximately 4%. A second NCO was assigned to assist in placing Main Feedwater in service. While placing No. 23 S/G onto Main Feedwater, No. 24 S/G level started decreasing below 25%. At 0220 hours, a reactor trip occurred on No. 24 S/G Low-Low Level. Following the trip, Auxiliary Feedwater was reduced and a main steamline isolation was

manually initiated to curtail a reactor cooldown following the trip. The cause of the No. 23 S/G High-High level signal and the reactor trip was Management/Quality Assurance Deficiency as related to maintenance activities on the No. 23BF19 valve (#23 S/G Feedwater Control Valve) during the recent refueling outage. Debris from maintenance activities was not removed from the inlet pipe to the valve. This debris kept the valve open during the startup. Corrective action included 23BF19 valve disassembly, cleaning and inspection; procedure modifications, positive disciplinary action, and management review of event with applicable personnel.

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, during startup, on No. 24 Steam Generator Low-Low Level

Event Date: 4/26/92

Report Date: 5/22/92

This report was initiated by Incident Report No. 92-293.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 2 Reactor Power 4%, startup in progress

DESCRIPTION OF OCCURRENCE:

On April 26, 1992, at 0219 hours, a Reactor Trip occurred on No. 24 Steam Generator (S/G) Low-Low Level {JC} during startup following the sixth refueling outage. This outage included extensive main turbine and generator repairs (reference LER 311/91-017-00).

Prior to the reactor trip, at 0153 hours, with reactor power at 3%, the No. 23 S/G level reached the High-High turbine trip/feedwater

isolation setpoint {JC}. At the time, feedwater supply to No. 23 S/G was in the process of switching to Main Feedwater {SJ}. No. 23 S/G was being fed by both Main Feedwater and Auxiliary Feedwater {BA}, while the other three (3) S/Gs were on Main Feedwater. As a result of the No. 23 S/G High-High level, a Turbine Trip signal and Main Feedwater isolation occurred per design. The Motor Driven Auxiliary Feedwater Pumps were running at the time of this event. Subsequently, the AF21 valves (S/G Auxiliary Feedwater Level Control Valves) were used to regulate feedwater flow to the four (4) S/Gs. Turbine Trip/Main Feedwater isolation on High-High S/G level is an Engineered Safety Feature (ESF). All associated components functioned per design.

Assessment of the cause of the High-High Level turbine trip signal, by the Senior Nuclear Shift Supervisor (SNSS), was that the Nuclear Control Operator (NCO) overfed No. 23 S/G.

As part of the preparation to relatch the No. 21 SGFP, the BF22 valves (S/G Feedwater Check Motor Operated Valves) are closed; however, the 22BF22 did not indicate close upon initiation of closure by the NCO.

DESCRIPTION OF OCCURRENCE: (cont'd)

The BF22 valves are not taken credit for as feedwater isolation valves. They are reverse flow check valves which have a limitorque motor operator to keep the valves closed during times when differential pressure is no longer available.

Approximately twelve (12) minutes after the High-High S/G Level signal, No. 21 SGFP was relatched and the startup resumed. Reactor power remained at approximately 4%. A second NCO was assigned to assist in placing Main Feedwater in service. Nos. 21 and 22 were placed on Main Feedwater. Following this, No. 24 S/G was placed on Main Feedwater (i.e., based on 24BF40 49% open indication).

While placing No. 23 S/G onto Main Feedwater, No. 24 S/G level started decreasing below 25%. The No. 24 S/G Feedwater Control Valve (24BF40) was opened further to stop No. 24 S/G's level decrease; however, this did not affect the rate of level decrease. At 0220 hours, a reactor trip occurred on No. 24 S/G Low-Low Level (i.e., 16%). Prior to the trip, reactor temperature remained constant at 547 degrees F.

Following the reactor trip, a cooldown occurred due to continued S/G feeding via the No. 21 SGFP and Auxiliary Feedwater. Initially Operations personnel reduced the SGFP speed; however, the cooldown continued. Subsequently, in accordance with Emergency Operating Procedure EOP-TRIP-2, Auxiliary Feedwater was reduced from 44E04 lb sub m/hr to 22E04 lb sub m/hr and Main Steamline Isolation (an ESF) was initiated stopping the cooldown.

Per the EOPs, the BF22 valves were closed; however, the No. 22BF22 valve again did not indicate closed. Subsequent investigation revealed that the valve's motor operator had failed. The valve had closed per design; although it did exhibit minor seat leakage.

Per Code of Federal Regulations 10CFR 50.72(b)(2)(ii), the Nuclear Regulatory Commission was notified of the No. 24 S/G Low-Low Level automatic Reactor Trip, No. 23 S/G High-High Level Turbine Trip/Feedwater Isolation signal, and the manually initiated post reactor trip Main Steamline Isolation.

Investigation of the above events identified a failed 23BF19 pilot valve and the auto/manual control station was found with failed capacitors. These caused sluggish valve response to both automatic and manual control demand signals. The pilot valve and the auto/manual control station were replaced. These failures were believed to have caused the No. 23 S/G High-High level and the No. 24 S/G Low-Low level trip signals. On May 4, 1992, at 0150 hours, plant startup was resumed. On May 5, 1992, upon reaching 20% reactor power, erratic S/G level indication was observed while initiating S/G Blowdown {WI}. Investigation revealed that the 23BF19 valve was stuck open approximately 0.25". Technical Specification 3.6.3 Action

DESCRIPTION OF OCCURRENCE: (cont'd)

"a" was entered and a controlled shutdown to Mode 2 (Critical) was initiated on May 5, 1992, at 0300 hours. At 0359 hours, Action "c" was met with the plant stable and in Mode 2.

Technical Specification 3.6.3 states:

"The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1."

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

"With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours."

APPARENT CAUSE OF OCCURRENCE:

Investigation of these events revealed that the operator did not overfeed No. 23 S/G causing the No. 23 S/G High-High level alarm as identified in the Description of Occurrence section. The cause of the No. 23 S/G High-High level signal and the reactor trip was Management/Quality Assurance Deficiency (as defined by NUREG 1022, "Licensee Event Report System) related to maintenance activities on the 23BF19 valve (No. 23 S/G Feedwater Control Valve) during the recently completed refueling outage. Debris from maintenance activities was not removed from the inlet pipe to the valve. This debris kept the valve open during the startup causing both the No. 23 S/G High-High level and the reactor trip. Excess flow to No. 23 S/G had caused reduced main feedflow to the other S/Gs.

During the outage, all four (4) BF19 valves were inspected as part of the preventive maintenance program. The BF19 valves are 14" air operated double seated globe valves. Disassembly of the 21, 22, and 24BF19 valves did not require special measures to perform the

APPARENT CAUSE OF OCCURRENCE: (cont'd)

inspections. However, the 23BF19 valve "cage", above the flow entry port, had to be cut out. Waste material, generated by the cutting procedures (i.e., plasma arc cutting, gouging, and grinding), was cleared from the valve lower portions but not from the inlet piping area prior to valve reassembly.

The BF19 valve work was conducted each shift over several days. No one supervisor was given responsibility to track the work effort. This resulted in ineffective control of relatched work activities and contingency planning.

Additional factors contributing to the inadequate cleaning/inspection of the 23BF19 valve were:

1. turnover requirements inadequate - turnover only consisted of log entries identifying completed work (without specific detail) and the next task to be performed; this contributed to inadequate closure inspections by Maintenance supervision and Quality Control personnel

2. procedure inadequacy - Maintenance Procedure SC.MD-PM.CN-0009(Q), "Steam Generator Feedwater Regulating Valve (BF-19) Internal Inspection", does not require closure inspection until after the plug has been installed thereby preventing inspection of the inlet piping and the surrounding cage area

A contributing factor to the cause of the No. 23 S/G High-High level and reactor trip events was the failed 23BF19 pilot valve and auto/manual station controller.

ANALYSIS OF OCCURRENCE:

The Low-Low S/G Level reactor trip prevents operation with S/G water level below the minimum required for adequate heat removal. The trip occurs on two out of three low-low level signals in any S/G. The setpoint ensures adequate S/G inventory, at the time of a reactor trip, to allow for possible Auxiliary Feedwater Pump starting delays; thus preventing S/G dryout and Reactor Coolant System {AB} thermal and hydraulic transients associated with a loss of the heat sink.

The BF19 valves regulate main feedwater flow to the S/Gs during normal power operation. During startup and low power operation the BF40 valves, in conjunction with the BF19 valves, regulate main feedwater flow.

The Reactor Protection System (RPS) {JC} functioned as designed and the heat sink was maintained during this event. Since the RPS is designed for the thermal and hydraulic effects of four-hundred (400) full power reactor trips, the Mode 2 reactor trip resulted in a

ANALYSIS OF OCCURRENCE: (cont'd)

thermal transient well within the design limits of the system. This event involved no undue risk to the health or safety of the public to the actuation of the RPS, Main Steamline Isolation, and the No. 23 S/G High-High level turbine trip/feedwater isolation signal, the circumstances surrounding these events are reportable in accordance with Code of Federal Regulations 10CFR 50.73 (a)(2)(iv).

CORRECTIVE ACTION:

Appropriate positive disciplinary action has been taken with maintenance personnel directly involved in the cleaning and inspection of the 23BF19 valve during the outage and with those operator(s) involved with the reactor startup following the No. 23 S/G High-High level signal.

Maintenance Department management has discussed the significance of the incident with those individuals involved in the BF19 outage work. Also, this will be discussed with all other applicable Maintenance Department personnel.

Procedure SC.MD-PM.CN-0009(Q) will be revised to include a precaution on debris intrusion protection. Also, the closure inspection requirement will be relocated to before installing the plug above the cage (noting that upstream and downstream piping require inspection).

Procedure SC.MD-GP.ZZ-0006(Q), "Tool and Miscellaneous Items, Cleanliness and Closure Control", will be revised to require its use for all closure work. The criteria for inspections will be listed.

Maintenance management will evaluate the need for additional administrative controls to support major work efforts.

The Control Room response to the Turbine Trip/Feedwater Isolation and Reactor Trip events has been reviewed with applicable Operations Department personnel.

The Nuclear Training Center will review the events addressed by this LER for addition to applicable Maintenance and Operations Department training programs.

The 23BF19 valve and associated piping was disassembled, cleaned and inspected prior to restoring it to service.

The feedwater control valves were tested, inspected and repaired as necessary.

The preventive maintenance program for the BF19, BF22 and BF40 valves will be reviewed and modified as required.

FAILURE DATA:

1. Valve - 23BF19

Manufacturer: Copes-Vulcan
Model Number: D-100-160-3
Failure: foreign material keeping valve open and faulty pilot valve

2. 22BF22 Valve Motor Operator

Manufacturer: Limitorque
Model Number: SMB-2
Failure - 22BF22 Failure to stroke

General Manager -
Salem Operations

MJP:pc

SORC Mtg. 92-062

PSE&G

Public Service Electric and Gas Company

P.O. Box 236 Hancocks Bridge, New Jersey 08038

Salem Generating Station

May 22, 1992

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 92-007-00

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 to be issued within thirty (30) days of event discovery.

Sincerely yours,

C. A. Vondra
General Manager -
Salem Operations

MJP:pc

Distribution

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