PSE&G

Public Service Electric and Gas Company 80 Park Place Newark, N.J. 07101 Phone 201/430-7000

March 27, 1980

Mr. Boyce H. Grier
Director of USNRC
Office of Inspection and Enforcement
Region 1
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. Grier:

LICENSE NO. DPR-70
DOCKET NO. 50-272
REPORTABLE OCCURRENCE 79-59/03L

Pursuant to the requirements of Salem Generating Station Unit No. 1 Technical Specifications, Section 6.9.1, we are submitting Licensee Event Report for Reportable Occurrence 79-59/03L. Following a review of 1979 Incident Reports, it was determined, in conjunction with the Resident Inspector, that the enclosed incidents are reportable as Licensee Event Reports.

Sincerely yours,

F. P. Librizzi
General Manager -
Electric Production

CC: Director, Office of Inspection
and Enforcement (30 copies)
Director, Office of Management
Information and Program Control
(3 copies)

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1903-1978
75
YEARS OF
SERVICE

8004010356

Report Number: 79-59/03L

Report Date: 3/27/80

Occurrence Date: April 24, May 8, June 30, 1979

Facility: Salem Generating Station
Public Service Electric & Gas Company
Hancock's Bridge, New Jersey 08038

IDENTIFICATION OF OCCURRENCE:

Loss of Residual Heat Removal (RHR) Pump

CONDITIONS PRIOR TO OCCURRENCE:

Refueling Mode 6

DESCRIPTION OF OCCURRENCE:

On April 24 and May 8, 1979, during Unit 1 first refueling outage, Relay Department personnel were performing tests on vital bus breakers. During the testing evolution, the vital bus supplying the operating RHR Pump inadvertently de-energized, causing loss of RHR flow. In each case, operating personnel were aware of the relay work and alerted to the possibility of loss of the operating RHR Pump. No core alterations or dilutions were in progress. RHR flow was restored within two minutes on April 24, 1979, and within four minutes on May 8, 1979.

On June 30, 1979, also during the refueling outage, the operating RHR Pump started to lose suction and was removed from service. The reactor vessel water level was checked at 97 feet 1 inch. The reactor vessel water level was increased to 97 feet 6 inches and the RHR Pump was returned to service. RHR flow was lost for 34 minutes.

DESIGNATION OF APPARENT CAUSE OF OCCURRENCE:

The cause of the occurrences on April 24, 1979 and May 8, 1979 was due to improper Relay Department action during the relay testing. On June 30, 1979, due to maintenance being performed on plant equipment, the reactor vessel water level was lowered to approximately one inch above the low operating level for the RHR Pump. This incident showed that the low operating level did not provide sufficient suction for the RHR Pump.

ANALYSIS OF OCCURRENCE:

Technical Specification 3.9.8 states, "At least one residual heat removal loop shall be in operation." Action Statement a. requires that "with less than one residual heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within four hours."

Action Statement b requires, "The residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of core alterations in the vicinity of the reactor vessel hot legs."

During all three occurrences, no core alteration or reactivity changes were in progress. The Technical Specification limiting condition for operation was not exceeded as the maximum time RHR flow was lost was 34 minutes.

As a point of interest, Amendment 24 to Salem Technical Specifications 3.4.1.1 in all modes except Refueling Modes, with reactor power below P-7 and with $K_{eff} < 1.0$, all Reactor Coolant Pumps and Residual Heat Removal Pumps may be de-energized for up to one hour, provided no operations are permitted which could cause dilution of the Reactor Coolant System boron concentration.

CORRECTIVE ACTION:

The Relay Department has reviewed and corrected their procedures and operating practices to prevent recurrence of this problem. An on-the-spot change has been implemented which changes the low reactor water level alarm set point from the centerline of the hot leg pipe (approximately 97 feet) to 97 feet 6 inches. These occurrences are considered isolated incidents and no further corrective action is required.

FAILURE DATA:

Not applicable

Prepared By A. W. Kapple

H. J. Midura 5/1/80
Manager - Salem Generating Station

SORC Meeting No. 20-80