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**Frederick W. Schneider**  
Vice President  
Production

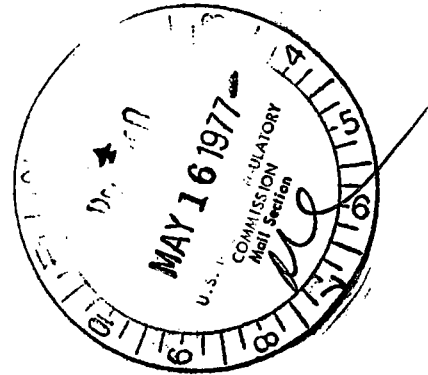
Public Service Electric and Gas Company 80 Park Place Newark, N.J. 07101 201/622-7000

May 10, 1977

Regulatory

File Cy.

Mr. James P. O'Reilly  
Director of USNRC  
Office of Inspections and Enforcements  
Region I  
631 Park Avenue  
King of Prussia, Pennsylvania 19406



Dear Mr. O'Reilly:

LICENSE NO. DPR-70  
DOCKET NO. 50-272

Pursuant to the requirements of Salem Generating Station Unit No. 1 Technical Specifications, Section 6.9.2, we are submitting ECCS Actuation Report No. 77-26/990. This report is required within ninety (90) days of the occurrence.

Sincerely yours,

Handwritten signature of Frederick W. Schneider in cursive.

771370186

Report Number: 77-26/990  
Report Date: 4/27/77  
Occurrence Dates: See Attachment 1  
Facility: Salem Generating Station  
Public Service Electric & Gas Company  
Hancocks Bridge, New Jersey 08038

Event

Appendix A Technical Specifications, Section 6.9.2 requires the reporting of Emergency Core Cooling System (ECCS) Actuations within 90 days of their occurrence. To date, we have experienced eight (8) such actuations. The purpose of this report is to describe the circumstances surrounding Safety Injection Nos. 7 and 8. Details of Safety Injections Nos. 1 thru 6 are contained in ECCS Actuation Report No. ECCS/77-01, previously submitted.

Discussion/Conclusion

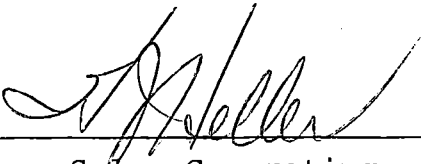
The referenced Westinghouse letter documents the acceptability of fifty (50) safety injection transients at a RWST temperature of 40°F. As the lowest RWST temperature in any of the subject transients was 61.5°F, none of the subject transients approaches the severity of the design basis transients and, as such, are acceptable.

References

- a) ECCS Actuation Report No. 77-26/990, Attachment 1
- b) Westinghouse Burl 3461 letter, dated 12/13/76, Attachment 2.

Prepared by T. L. Spencer

SORC Meeting No. 46-77

  
\_\_\_\_\_  
Manager - Salem Generating

SAFETY INJECTION NO. 7

At 1308 hours on 3/30/77, a Safety Injection/Reactor and Turbine Trip signal was initiated due to Steam Differential Pressure Low Pl. Prior to this event, the unit was in Mode 1 at 75% reactor power, 880 MWe generator load. Operations Department surveillance procedures SP(O) 4.3.2.1.1(E) for Technical Specification surveillance 4.3.2.1.1 was in progress. Step 5.2.34A states "Open and then close the following valves to clear the status panel indicating lights: 13MS7, 14MS7, 13MS18, 14MS18". The bezel from which these valves are operated also contains the open/close control pushbuttons for 13MS167 and 14MS167 (13 and 14 Main Steam Isolation Valves). Attempting to perform step 5.2.34A, an operator mistakenly pushed the close pushbuttons for 13 and 14MS167. The closing of the 13 and 14 MSIV's caused the Safety Injection. The cause of this event was operator error.

SAFETY INJECTION NO. 8

At 1001 hours on 4/12/77, a loss of the 1G 4kV Group Bus caused the loss of No. 14 RCP. Plant conditions were stabilized at 25% reactor power, Tave at 566°F, Rod Control System in manual, three (3) loop operation. (ref. LER 77-25/01T). At 1032 hours, the 1G Group Bus had been re-energized and an attempt was made to restart No. 14 RCP. The resultant S/G water level transient caused a No. 14 S/G Hi-Hi level Turbine/Reactor Trip. Immediately following the trip, a Safety Injection occurred.

With Tave at 566°F, a turbine trip caused the Steam Dump - Turbine Trip Controller to initiate a Hi Tave steam dump sequence. The steam dump sequence resulted in six (6) steam dump valves opening fully and three (3) of the remaining six (6) to modulate to a 1/2 open position. The resultant steam flow through the steam dump valves combined with a 4-5% flow spike was sufficient to cause Nos. 11, 12 and 13 S/G Hi Steam Flow alarms to actuate. When steam dump was initiated, Tave decreased to less than 543°F causing a low Tave signal. The cause of the Safety Injection was Hi Steam Flow coincident with low Tave.

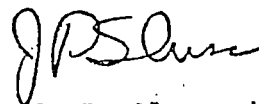
R. D. Rippe

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BURL-3461

Thus in spite of the original Salem design basis using 31.1 piping codes which did not specifically require transient design calculations for the subject transient, we believe that our more recent analysis provides a sound basis for acceptability of the Salem piping.

Very truly yours,  
WESTINGHOUSE ELECTRIC CORPORATION



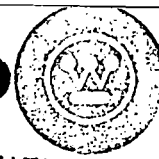
J. P. Sluss, Manager  
Salem Project

/hs

cc: R. D. Rippe, 3L  
D. J. Jagt, 1L  
C. F. Barclay, 1L  
J. J. Dolan, 1L

5000 Y 110 NY  
EAL TNT, BG  
CHIEF CONTROLS Engg  
MGR - SALEM  
Westinghouse Electric Corporation

Attachment No. 2



BURL-3461

SALEM GEN. STATION

Power Systems

21 DEC 76

11

Water Reactor Divisions

NGR *SA* P.E. *SA* C.E. *SA*

Box 355

Pittsburgh Pennsylvania 15230

O.A.S. *SA* M.E. *SA* SOAE *SA*

SECURITY

*file*

DECEMBER 13, 1976

OE	<i>ZAC</i>	SE	<i>SA</i>
		RE	
HP	<i>SA</i>		

Mr. R. D. Rippe  
Chief Mechanical Engineer  
Electric Engineering Department  
Public Service Electric & Gas Company  
60 Park Place  
Newark, New Jersey 07101

CHIEF MECHANICAL ENGINEER ENGINEERING DEPT.		
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DEC 16 1976		
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Dear Mr. Rippe:

SALEM NUCLEAR GENERATING STATION  
UNITS NUMBER 1 AND 2  
Safety Injection (SI) Transient  
Design Basis

During the recent pre-critical testing phase, the plant was subjected to three (3) inadvertent Safety Injection (SI) initiation events, which we understand resulted in some water being injected into the Reactor Coolant Loop. We also understand that the NRC has verbally asked for the design transient basis for Salem for this type of event.

While we have not specifically analyzed Salem for this type of transient, we are confident that our ongoing plant analysis associated with ASME Section III more than demonstrates that the recent three (3) SI's will have no detrimental effect on Salem. Our conclusion is based on the following rationale.

We have analyzed sufficient Section III piping systems including piping similar to yours with the 1-1/2" SI nozzles to show that fifty (50) such SI events can be accommodated without exceeding the appropriate stress limits at the SI nozzle.

These analysis were based on the nozzles being subjected to a 40°F water transient which is probably far worse than the actual transient seen at Salem.

The results of these analysis are in the process of being formalized for submittals to the NRC for Section III plants.

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