

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Salem Generating Station - Unit 1										DOCKET NUMBER (2) 0 5 0 0 0 2 7 2										PAGE (3) 1 OF 05																					
TITLE (4) Containment Systems - Type B and C Leak Rate - Out of Specification																																									
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 3			0 6			8 4			8 4			0 0			7			0 1			0 3			1 9			8 5									0 5 0 0 0					
OPERATING MODE (9) 5						THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)																																			
POWER LEVEL (10) 01010						20.402(b)						20.406(e)						50.73(a)(2)(iv)						73.71(b)																	
						20.406(a)(1)(i)						50.36(e)(1)						50.73(a)(2)(v)						73.71(e)																	
						20.406(a)(1)(ii)						50.36(e)(2)						50.73(a)(2)(vi)						OTHER (Specify in Abstract below and in Text, NRC Form 365A)																	
						20.406(a)(1)(iii)						50.73(a)(2)(i)						50.73(a)(2)(viii)(A)																							
						20.406(a)(1)(iv)						50.73(a)(2)(ii)						50.73(a)(2)(viii)(B)																							
20.406(a)(1)(v)						50.73(a)(2)(iii)						50.73(a)(2)(x)																													
LICENSEE CONTACT FOR THIS LER (12)																																									
NAME J. L. Rupp																				TELEPHONE NUMBER AREA CODE 6 0 9 3 3 9 - 4 3 0 9																					
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)																																									
CAUSE			SYSTEM			COMPONENT			MANUFACTURER			REPORTABLE TO NRC						CAUSE			SYSTEM			COMPONENT			MANUFACTURER			REPORTABLE TO NRC											
X			LIF			V I I I			V I O I			8 5			Y																										
SUPPLEMENTAL REPORT EXPECTED (14)																				EXPECTED SUBMISSION DATE (15)										MONTH DAY YEAR											
YES (If yes, complete EXPECTED SUBMISSION DATE)																				NO																					

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On March 6, 1984, during a refueling outage, PSE&G commenced routine periodic leak rate testing of Type C components; at which time, a check valve (1SA119) located inside containment and in the containment Station Air supply line, exhibited a leakage rate greater than the capacity of the leak rate test equipment. Because of this, the actual leak rate and the total combined leakage of all Type B and C components could not be verified to be within specifications. Leak rate testing continued and, although the leak rate of several valves exceeded administrative limits, total leakage for each penetration was within allowable limits. All valves involved were repaired and satisfactorily tested. On August 24, 1984, the final results of Type B and C leak rate testing demonstrated that combined leakage was within specification.

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PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are indentified in the text as [XX].

IDENTIFICATION OF OCCURRENCE:

Containment Systems - Type B and C Leak Rate - Out-of-Specification

Event Date: 03/06/84

Report Date: 03/19/85

This report was initiated by Incident Report No. **84-040**

CONDITIONS PRIOR TO OCCURRENCE:

Mode 5 - Rx Power 000 % - Unit Load 0000 MWe

DESCRIPTION OF OCCURRENCE:

Leak rate testing of Type B and C components commenced on March 6, 1984, during a scheduled refueling outage, at which time, 1SA119 exhibited a leak rate greater than the maximum range of the leak rate test equipment (20,000 sccm). 1SA119 is a check valve (located inside of containment) in the Station Air [LF] supply line to the containment. 1SA118 (in series with 1SA119, and located outside of containment) is the Containment Station Air Supply Manual Isolation Valve. Leak rate testing of 1SA118 indicated that the leakage (through 1SA118) was within the limits allowed by the Technical Specifications. 1SA118 is a normally closed valve; opened only when Station Air is required inside of containment. However, 1SA118 was opened (to supply air to the containment) at various times during operation in Modes 1 through 4. Because of this, and the fact that the actual leak rate of 1SA119 could not be ascertained, it was not possible to demonstrate that the total Type C leakage was less than 0.60 La as required by Technical Specifications Limiting Condition for Operation 3.6.1.2.b (applicable only in Modes 1 through 4), during the periods when 1SA118 was opened. The findings were therefore reportable, and were documented in the original report (Unit 1 LER 84-007-00), dated April 5, 1984.

Testing continued, and on March 7, 1984, Steam Generator Blowdown Isolation Valves 11 through 14GB4 were tested. Testing results indicated that the leakage through these valves was greater than the assigned administrative limits.

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DESCRIPTION OF OCCURRENCE: (cont'd)

On March 14, 1984, 12VC19 (located inside the containment in the post LOCA air sampling line penetration) was tested and found to have a leak rate of greater than the maximum range of the leak rate test equipment (20,000 sccm). The redundant valve in this penetration (12VC17), located outside of the containment, was tested with satisfactory results.

On May 24, 1984, 1WL96 (located inside the containment in the Reactor Coolant Drain Tank gas analyzer sample line penetration) was tested and found to have a leak exceeding the assigned administrative limit. The redundant valve in this penetration (1WL97), located outside of the containment, was tested with satisfactory results.

On July 2, 1984, 1WR80 (located inside the containment in the primary makeup water supply line penetration) was tested and found to have a leak rate exceeding the assigned administrative limit. The redundant valve in this penetration (1WR81), located outside of the containment, was tested with satisfactory results.

All valves involved were repaired and satisfactorily tested. Testing was completed on August 24, 1984; combined Type B and C leakage was determined to be within specification prior to returning to Mode 4 operation.

APPARENT CAUSE OF OCCURRENCE:

The seat of check valve 1SA119 was found to contain some corrosion. Disassembly of 11 through 14GB4 valves revealed that the seating surfaces were damaged by wire cutting. 12VC19 was discovered to have a packing leak. Inspection of the valve seats of 1WL96 and 1WR80 revealed some slight irregularities which could be removed by valve lapping.

ANALYSIS OF OCCURRENCE:

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of forty-seven (47) psig.

Technical Specification Action Statement 3.6.1.2 requires:

With either the measured overall integrated containment leakage rate exceeding 0.75 La, or with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 La, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System [AB] temperature above two-hundred (200) degrees.

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ANALYSIS OF OCCURRENCE: (cont'd)

Leakage rate limits during the performance of periodic tests incorporate conservatism to account for possible degradation of the leakage barriers between testing. The combined leakage rate of all Type B and C components had previously been demonstrated within specification during the last leak rate testing performed during the period of October 1982 to February 1983.

In the case of 11 through 14GB4 valves, the steam generator tubes constitute a valid primary system pressure boundary. Because No. 11, 12, 13 and 14 Steam Generators have no tube leaks, the seat leakage through the GB4 valves had no effect on containment leak rates. The penetrations containing 12VC19, 1WL96 and 1WR80 all contained redundant isolation valves, located outside of the containment, which satisfactorily passed the first leak rate test. In addition, 12VC19 leakage was attributed to packing leakage; since this valve was located in the containment, leakage would be confined to the containment atmosphere.

Although 1SA118 was opened at various times during operation in Modes 1 through 4, Station Air pressure is approximately 100 psig to 110 psig. Containment pressure is normally maintained between minus 0.1 and plus 0.3 psig. Even during an accident condition, maximum containment pressure would be 47 psig, which is well below normal Station Air pressure. Therefore, the possibility of leakage out of this connection is minimum, and the event involved no undue risk to the health or safety of the public. However, since the actual leak rate of 1SA119 could not be ascertained, it was not possible to demonstrate that the total Type C leak rate was less than 0.60 La as required by Technical Specifications Limiting Condition for Operation 3.6.1.2.b (applicable only in Modes 1 through 4), during the periods when 1SA118 was opened. Because of this, the event is reportable in accordance with the Code of Federal Regulations, 10CFR 50.73(a)(2)(ii).

CORRECTIVE ACTION:

The seat on the flapper from check valve 1SA119 was removed and machined. The valve seat was lapped, and the valve was reassembled. 1SA119 was retested on July 8, 1984, with satisfactory results. The internals (plug, stem and seat) from 11, 13 and 14GB4 valves were replaced with new components. The plug from 12GB4 was machined, and the valve seat was lapped. The GB4 valves were retested as follows: 13GB4 on June 20, 12GB4 on June 23, 11GB4 on June 24 and 14GB4 on June 29, 1984; all tests were satisfactory. The packing on 12VC19 was adjusted, and the valve was immediately retested on March 14, 1984; the test results were satisfactory. Valves 1WL96 and 1WR80 were lapped. 1WL96 was retested on June 28, and 1WR80 was retested on July 3, 1984; both retests were satisfactory. On August 24, 1984, the final results of Type B and C leak rate testing demonstrated combined leakage was within specification.

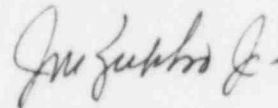
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CORRECTIVE ACTION: (cont'd)

At the present time, Technical Specifications do not list 1SAll9 as a containment isolation valve which requires periodic leak rate testing. As a conservative measure, Field Directive S-C-A900-MFD-068 (issued prior to this occurrence) identifies this valve (among others) as a containment isolation valve, and directs the leak rate testing to be performed. In the future, a License Change Request will be submitted, requesting that these valves be added to the Technical Specification list of containment isolation valves. Until then, PSE&G will consider this valve as a containment isolation valve.

The GB4 valves are presently included in the Type C leak rate testing program. However, in the future, these valves will be deleted from the Type C leak rate testing program, and as an alternative, will be subjected to a leak rate test more appropriate to the function that these valves perform in mitigating the consequences of a steam generator tube rupture. In the interim, the GB4 valves shall continue to be subject to Type C leak rate testing until the requested Technical Specification Table 3.6-1 revision has been approved, and the Technical Specification has been amended to remove the valves from the Type C test requirements.



General Manager-
Salem Operations

JLR:tns

SORC Mtg 85-037



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

Salem Generating Station

March 19, 1985

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

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-70
DOCKET NO. 50-272
UNIT NO. 1
LICENSEE EVENT REPORT 84-007-01
SUPPLEMENTAL REPORT

Pursuant to the requirements of 10CFR 50.73(a)(2)(ii), we are submitting supplemental Licensee Event Report for Reportable Occurrence 84-007-01.

Sincerely yours,

A handwritten signature in dark ink, appearing to read "J. M. Zupko, Jr.", written in a cursive style.

J. M. Zupko, Jr.
General Manager -
Salem Operations

JR:tcs

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