NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Salem Generating Station - Unit 1	0 5 0 0 0 2 7 2	1 OF 0 4

TITLE (4)
Containment Isolation Valves 1CV68 and 1CV69 Inoperable

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)
1 0	1 9	8 4	8 4	- 0 2 2	- 0 0	1 1	1 6	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
2	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
0 0 0	20.405(a)(1)(ii)	X 50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
	20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
	20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
J. L. Rupp	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 NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	
B	C B	I S V	V 0 8 5	Y							
B	C B	I S V	V 0 8 5	Y							

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

Between 2130 hours, October 18, 1984, and 1000 hours, October 19, 1984, while performing maintenance on 1CV68 (Charging Header Stop Valve), the redundant stop valve (1CV69) was relied upon to satisfy the containment isolation valve requirements of Technical Specification 3.6.3.1. However, unknown to the operating shift, 1CV69 had failed the Type "C" leak rate test. Upon discovery, corrective actions were immediately taken to isolate the penetration until 1CV68 could be returned to an operable status. When the Type "C" leak rate testing was performed (while the Unit was in Mode 5), 1CV68 passed. This satisfied the requirements for entry into Mode 4; however, due to an oversight, the operators were not informed of the unsatisfactory testing results of 1CV69. Consequently, when a packing leak developed on 1CV68, 1CV69 was inappropriately relied upon for containment isolation purposes. This is the only penetration containing two valves, either of which will satisfy the Technical Specification requirement; therefore, a Type "C" leak rate procedural change will inform the Operations Department of unsatisfactory leak rate tests on either valve, thus preventing recurrence. Due to not meeting the time requirements for a limiting condition for operation, this event is reportable in accordance with 10CFR 50.36(c)(2).

IE22
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NRC Form 366
(9-83)

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

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

IDENTIFICATION OF OCCURRENCE:

Containment Isolation Valves 1CV68 and 1CV69 Inoperable

Discovery Date: 10/19/84

Report Date: 11/16/84

This report was initiated by Incident Report No. **84-163**

CONDITIONS PRIOR TO OCCURRENCE:

Mode 2 - Rx Power 000 % - Unit Load 0000 MWe

DESCRIPTION OF OCCURRENCE:

At approximately 2130 hours, October 18, 1984, a retest of 1CV68 (Charging Header Stop Valve), following valve repacking, was performed with unsatisfactory results. Technical Specification Table 3.6-1 requires either 1CV68 or the redundant stop valve (1CV69) to be operable. A satisfactory 4.0.5-V test was performed on 1CV69 to prove operability, and maintenance continued on 1CV68.

At 1000 hours, October 19, 1984, it was learned that 1CV69 could not be considered an operable containment isolation valve because it had recently failed to pass a satisfactory Type "C" leak rate test. Technical Specification Action Statement 3.6.3.1.b was entered at this time, and 1CV69 was deactivated and tagged shut in accordance with the action requirements. Realizing that the time requirements of Technical Specification Limiting Condition For Operation 3.6.3.1 were not met (because both 1CV68 and 1CV69 were inoperable on October 18, 1984, and no action was taken until October 19, 1984), in compliance with the Code of Federal Regulations, 10CFR 50.36(c)(2), the Nuclear Regulatory Commission was notified of the event at 1150 hours, October 19, 1984.

1CV68 was subsequently repaired, satisfactorily tested and returned to an operable status. Technical Specification Action Statement 3.6.3.1.b was terminated at 1619 hours, October 19, 1984.

APPARENT CAUSE OF OCCURRENCE:

Technical Specification 3.6.3.1 requires the containment isolation valves listed in Table 3.6-1 to be operable in Modes 1, 2, 3 and 4. As previously stated, the operability of either 1CV68 or 1CV69 will satisfy the Technical Specification requirement for that particular penetration. It should be noted that this is the only containment penetration of its kind; i.e., the only one containing two valves, either of which will satisfy this Technical Specification requirement.

The Type "C" leak rate testing of containment isolation valves was performed while the Unit was in Mode 5. Had any of the valves (other than 1CV68 or 1CV69) failed to satisfactorily pass the test, entry into Mode 4 would not have been authorized. The testing results identified 1CV69 as unsatisfactory; however, 1CV68 satisfactorily passed the test which satisfied the Technical Specification requirement for entering Mode 4. Knowing that the Mode 4 Technical Specification requirements for containment isolation valves had been satisfied, and not anticipating a problem with 1CV68, the operators were not notified of the unsatisfactory leak rate test on 1CV69. Due to this oversight, when a packing leak developed on 1CV68, 1CV69 was inappropriately relied upon for automatic containment isolation purposes.

ANALYSIS OF OCCURRENCE:

The operability of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment, in the event of a release of radioactive material to the containment atmosphere or from pressurization of the containment. Containment isolation, within the time limits specified in the Technical Specifications, ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a Loss of Coolant Accident (LOCA).

In the event that a LOCA had occurred during the time period when 1CV68 was inoperable for repairs and 1CV69 was technically inoperable, a containment isolation signal would have closed 1CV69. At the same time, the charging pumps would have automatically started. Although leak rate testing indicated unacceptable seat leakage through 1CV69, and since the discharge pressure of the charging pumps is not isolated from 1CV69, the result would have been containment in-leakage rather than containment out-leakage.

This occurrence did not affect the health or safety of the public. However, due to not meeting a limiting condition for operation, the event is reportable in accordance with the Code of Federal Regulations, 10CFR 50.36(c)(2).

CORRECTIVE ACTION:

As previously stated, when it was discovered that 1CV69 did not satisfactorily pass the Type "C" leak rate test, and therefore could not be considered an operable containment isolation valve, the appropriate Technical Specification Action Statement was entered, the valve was deactivated and tagged closed and the Nuclear Regulatory Commission was notified of the event. 1CV68 was repaired, tested, declared operable and the Action Statement was terminated.

To prevent recurrence, the Type "C" Leak Rate Testing Procedure will be changed to ensure that the Operations Department is notified in the event of unsatisfactory leak rate tests on either 1CV68 or 1CV69.

General Manager-
Salem Operations

JLR:tns

SORC Mtg 84-154

PSE&G

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

Salem Generating Station

November 16, 1984

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 84-022-00

This Licensee Event Report is being submitted pursuant to the requirements of 10CFR 50.36(c)(2). This report is required within thirty (30) days of discovery.

Sincerely yours,

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

JR:k11

CC: Distribution

IE22
1/1