NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 | V A S P S 2 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | | (5)

LICENSEE CODE 9–14 | LICENSE NUMBER 15–25 | LICENSE TYPE 26–30 | CAT 57–58

CON'T 0 1 REPORT SOURCE L (6) | 0 5 0 0 0 2 8 0 (7) | 0 8 0 3 8 3 (8) | 0 9 0 2 8 3 (9)

DOCKET NUMBER 61–68 | EVENT DATE 69–74 | REPORT DATE 75–80

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 Type "C" testing performed during the 1983 refueling outage for Unit No. 2

0 3 disclosed an "as found" leakage exceeding the allowable leakage rate delineated in

0 4 T.S.-4.4.C. This is contrary to T.S.-3.8.A.1 and is reportable per T.S,6.6.2.b.(2).

0 5 Actual containment leak rate would have been considerably less than the value

0 6 leak rate computed for the test. Therefore, the health and safety of the public

0 7 would not have been affected.

0 8

SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
S D (11)	E (12)	B (13)	V A L V E X (14)	X (15)	X (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 3	—	0 3 3	/	0 3	L	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	X (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	N (24)	A (25)	X 9 9 9 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The major cause of leakage was determined to be deterioration and wear of the

1 1 valves due to normal usage. The valves have been repaired and retested with

1 2 satisfactory results.

1 3

1 4

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	H (28)	0 0 0 (29)	N/A	B (31)	Type "C" testing

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	N/A	N/A

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 0 (37)	Z (38)	N/A

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	N/A

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	N/A

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
2 0	N (44)		

8309090395 830902
PDR ADOCK 05000281
S PDR

NAME OF PREPARER J. L. Wilson PHONE: (804) 357-3184

ATTACHMENT 1
SURRY POWER STATION, UNIT NO. 2
DOCKET NO: 50-281
REPORT NO: 83-033/03L-0
EVENT DATE: 08-03-83

TITLE OF THE EVENT: CONTAINMENT ISOLATION VALVES LEAKAGE

1. Description of the Event

With the Unit in a Refueling outage, it was discovered through performance of Periodic Test 16.4 (Containment Isolation Valve Leakage, Type 'C' testing) that the "As Found" leakage exceeded the allowable leakage specified in Technical Specification 4.4.C. This is contrary to Tech. Spec. 3.8.A.1 and is reportable in accordance with Tech. Spec. 6.6.2.b.(2).

2. Probable Consequences and Status of Redundant Equipment

Total leakage for the test is computed by summing the leak rates of all the valves tested. This very conservative method does not take into consideration two or more isolation valves in series for the same containment penetration. Actual leakage through the penetration would be that of the valve with the smallest leak rate. Since the actual containment leak rate would have been considerably less than the valve leak rate computed by the test, the health and safety of the public would not have been affected.

3. Cause

The major cause of leakage was determined to be deterioration and wear of the valves due to normal usage.

4. Immediate Corrective Action

As each valve with excessive leakage was identified, maintenance activities were initiated to have the valve repaired.

5. Subsequent Corrective Action

The affected valves have or will be repaired and subsequent testing will verify that the total combined leakage for the valves is within specification.

6. Action Taken to Prevent Recurrence

The containment leakage will continue to be monitored as required by 10CFR 50, Appendix J. In addition, a Type A test will be performed to verify containment integrity. An engineering review of the testing data and equipment selection will be conducted to determine possible corrective actions/modifications.

7. Generic Implications

None.

Vepco

VIRGINIA ELECTRIC AND POWER COMPANY
Surry Power Station
P. O. Box 315
Surry, Virginia 23883

Serial No: 83-061

SEP 2 1983

Docket No: 50-281

License No: DPR-37

Mr. James P. O'Reilly
Regional Administrator
Suite 2900
101 Marietta Street, NW
Atlanta, Georgia 30303

Dear Mr. O'Reilly

Pursuant to Surry Power Station Technical Specifications, the Virginia Electric and Power Company hereby submits the following Licensee Event Report for Surry Unit 2.

Report Number	Applicable Technical Specification
83-033/03L-0	T. S. 6.6.2.b(2)

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be reviewed by Safety Evaluation and Control.

Very truly yours,

J. L. Wilson
Station Manager

Enclosure

cc: Document Control Desk, USNRC
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Washington, D. C. 20555