10CFR50.73

Virginia Electric And Power Company
Surry Power Station
5570 Hog Island Road
Surry, Virginia 23883

February 25, 2000

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Serial No.: 00-082
SPS: BAG
Docket No.: 50-281
License No.: DPR-37

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 2.

Report No. 50-281/2000-001-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

E. S. Grecheck, Site Vice President
Surry Power Station

Enclosure

Commitments contained in this letter:

1. Upon completion of the root cause evaluation, the approved recommendations will be implemented through the Corrective Action Program.

IE22

cc: U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23 T85
Atlanta, Georgia 30303-8931

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 06/30/2001

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
SURRY POWER STATION , Unit 2	05000 – 281	1 OF 4

TITLE (4)

Containment Isolation Valve Found with Unacceptable Leakage During Maintenance

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 NAME	DOCUMENT NUMBER 05000-
01	27	00	00	01	00				FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	100%		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)	X	50.73(a)(2)(v)		Specify in Abstract below
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
E. S. Grecheck, Site Vice President	(757) 365-2001

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	JM	ISV	Crosby	Y					

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

On January 27, 2000, a review of testing results indicated that during the replacement of 2-DA-TV-200A, the leakage through 2-DA-TV-200B was in excess of the allowable Type "C" limits. On January 25, 2000, with Unit 2 at 100% reactor power, 2-DA-TV-200A was replaced in less than one hour. The excessive leakage of 2-DA-TV-200B was not discovered until performing the post maintenance testing of 2-DA-TV-200A. Once this condition was identified, appropriate actions were taken in accordance with Technical Specifications. 2-DA-TV-200B was replaced, satisfactorily tested, and returned to service. Based upon preliminary indications, it is believed that 2-DA-TV-200B failed because dirt and debris became impacted between the ball and valve seat causing excessive wear. A root cause evaluation has been initiated and approved recommendations will be implemented through the Corrective Action Program.

No conditions adverse to safety resulted from this event and the health and safety of the public were not affected. A 4-hour report was made on January 27, 2000 pursuant to 10CFR50.72(b)(2)(iii)(C). This event is being reported pursuant to 10CFR50.73(a)(2)(v)(C) and 10CFR50.73(a)(2)(i)(B).

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TEXT (If more space is required, use additional copies of NRC Form 366A) **(17)**

1.0 DESCRIPTION OF THE EVENT

On January 20, 2000, a Unit 2 inside containment isolation valve, 2-DA-TV-200A [EIIS-JM,ISV], was cycled and the stroke time was measured to comply with the quarterly testing requirements of the Inservice Testing (IST) Program. At 1339 hours, the stroke time for 2-DA-TV-200A was measured at 2.59 seconds, exceeding the IST maximum stroke time of 2.0 seconds. 2-DA-TV-200A was declared inoperable. At 1500 hours, the corresponding outside containment isolation valve, 2-DA-TV-200B [EIIS-JM,ISV], was closed and deactivated in accordance with the requirements of Technical Specification (TS) 3.8.C.1.b.

On January 25, 2000, at 1006 hours, with Unit 2 at 100% reactor power, a team entered containment to replace 2-DA-TV-200A. The valve was replaced and at 1055 hours, 2-DA-TV-200A was stroked satisfactorily. 2-DA-TV-200A was removed from the system for less than 49 minutes.

To complete post maintenance testing (PMT) of 2-DA-TV-200A, a 10CFR50 Appendix J Type "C" containment leak rate test was performed on January 25, 2000, at 1549 hours. Test results indicated that leakage was greater than 25 standard cubic feet per hour (scfh). However, because the test air was supplied between 2-DA-TV-200A and 2-DA-TV-200B, it could not be determined which valve was leaking. The leakage was assumed to be through 2-DA-TV-200A based on a history of successful as-found Type "C" testing on 2-DA-TV-200B and the possibility that 2-DA-TV-200A became misadjusted during installation.

On January 26, 2000, the valve alignment used during the leak rate test was changed to verify that the excessive leakage was through 2-DA-TV-200A. At 1338 hours, it was determined that the leakage through 2-DA-TV-200A was less than 0.07 scfh. The valve alignment was then changed to allow leak testing of 2-DA-TV-200B and, at 1353 hours, leakage through 2-DA-TV-200B was determined to be 78.86 scfh (although not considered accurate due to the line not being verified as drained). Due to the as-found leakage on 2-DA-TV-200B, a 1-hour TS action statement was entered as a conservative measure to establish containment integrity in accordance with TS 3.8.A.1 (subsequent review indicates that a 4-hour action statement in accordance with TS 3.8.C.1.b would have been acceptable). At 1405 hours, 2-DA-TV-200A was verified closed and deactivated in accordance with the requirements of TS 3.8.C.1.b.

On January 27, 2000 at 1201 hours, a Type "C" containment leak rate test performed (with the line verified to be drained) on 2-DA-TV-200B determined the as-found leakage to be 182.57 scfh. The 10CFR50 Appendix J Type "B" & "C" total leakage acceptance criteria is 180 scfh.

A review of the subject maintenance and testing activities revealed that 2-DA-TV-200B had been inoperable during the replacement of 2-DA-TV-200A. This condition alone could have prevented the fulfillment of the safety function of structures or systems that are needed to control the release of radioactive material. A 4-hour non-emergency report to the NRC Operations Center was made on January 27, 2000 at 1538 hours pursuant to 10CFR50.72(b)(2)(iii)(C). This report is being made pursuant to 10CFR50.73(a)(2)(v)(C).

Based upon the available information regarding valve leakage and stroke times, appropriate actions were taken to ensure compliance with TSs. However, a review of the testing results revealed that 2-DA-TV-200A should have been closed and deactivated on January 25, 2000, to ensure containment integrity. Therefore, this report is also being made pursuant to 10CFR50.73(a)(2)(i)(B), as an operation or condition prohibited by TSs since 2-DA-TV-200A was closed but not deactivated.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

The reactor containment sump pump discharges through two normally open pneumatically operated trip valves, 2-DA-TV-200A and 2-DA-TV-200B, located on each side of the containment wall. These valves function as part of the Reactor Containment Isolation System [EIIS-JM]. The valves close upon receipt of a containment isolation signal (safety injection). Containment isolation valves are tested with design pressure to ensure isolation during a design basis accident. The measured leakage under test conditions through 2-DA-TV-200B was 182.57 scfh. The measured leakage under test conditions from other pathways was 142.08 scfh. The total leakage from all pathways during the period of concern (i.e., January 25, 2000, for less than 1 hour when 2-DA-TV-200A was removed and 2-DA-TV-200B exceeded acceptable leakage rates) exceeded an allowable limit of 305 scfh.

Considering the conditions discovered during this event, a review of the model used to assess off-site doses from the release of radionuclides for a design basis Loss of Coolant Accident (LOCA) at Surry was performed. The review concluded that with the measured leakage from this event, the calculated off-site doses remained within the regulatory limits stated in 10 CFR 100. The review also concluded that the potential control room doses would have remained within the limits specified by General Design Criteria 19.

The time period in which potential increased leakage through the containment sump pump discharge line existed was less than one hour (which is the allowed out of service time for containment integrity in accordance with TS 3.8.A.1.a). The probability of a LOCA occurring during this event remained small at approximately 5.0E-8.

In conclusion, this event resulted in no safety consequences or significant implications, and the health and safety of the public were not affected.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

3.0 CAUSE

Based upon preliminary indications, it is believed that 2-DA-TV-200B failed because dirt and debris became impacted between the ball and valve seat causing excessive wear. A root cause evaluation (RCE) has been initiated and will determine the cause of the failure.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

On January 26, 2000, at 1405 hours, after leak rate results on valve 2-DA-TV-200B were found to exceed the acceptance criteria, 2-DA-TV-200A was verified closed and deactivated in accordance with TS 3.8.C.1.b.

5.0 ADDITIONAL CORRECTIVE ACTIONS

Valve 2-DA-TV-200B was replaced, tested satisfactorily, and returned to service.

6.0 ACTIONS TO PREVENT RECURRENCE

An RCE has been initiated and, upon completion, the approved recommendations from the RCE will be implemented through the Corrective Action Program.

7.0 SIMILAR EVENTS

None

8.0 MANUFACTURER/MODEL NUMBER

Crosby 2 inch, 150 pound ball valve with Bettis air actuator NCB 415-SR80, Drawing # D-SC-96162

9.0 ADDITIONAL INFORMATION

Unit 1 was operating at 100% reactor power and was not affected by this event.