10CFR50.73

Virginia Electric and Power Company
North Anna Power Station
P. O. Box 402
Mineral, Virginia 23117

February 16, 2001

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

Serial No.:	01-054
NAPS:	MPW
Docket No.:	50-339
License No.:	NPF-7

Dear Sirs:

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

Report No. 50-339/2001-001-00

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,

D. A. Heacock
Site Vice President

Commitments contained in this letter: None

Enclosure

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

Mr. M. J. Morgan
NRC Senior Resident Inspector
North Anna Power Station

IE22

NRC FORM 366 (1-2001)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
NORTH ANNA POWER STATION UNIT 2	05000 339	1 OF 4

TITLE (4)

UNIT SHUTDOWN REQUIRED DUE TO IDENTIFIED REACTOR COOLANT LEAKAGE EXCEEDING LIMITS

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	19	2001	2001	01	00	02	16	2001	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	99

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)	50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)	73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)	73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)	OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)	
	20.2203(a)(2)(iv)	x	50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)	
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)	
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)	
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
David A. Heacock	(540) 894-2101

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	AB	ISV	R677	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 15 single-spaced typewritten lines) (16)

On January 19, 2001, at approximately 1145 hours, with Unit 2 in Mode 1 at 99 percent power, a Notification of Unusual Event (NOUE) was declared, as a result of ramping Unit 2 offline due to identified Reactor Coolant System leakage greater than the Technical Specifications (TS) limit of 10 gallons per minute (gpm). Actual leakage was measured at 10.0015 gpm. At 1215 hours, a 1 hour report was made to the NRC Operations Center in accordance with 10 CFR 50.72(a)(1)(i). Unit 2 was ramped offline in accordance with the TS and placed in hot standby. The source of the identified leakage was determined to be from the valve stem **stuffing box** on the C reactor coolant loop bypass valve. The valve was placed on its backseat and leakage stopped at 2145 hours. The C reactor coolant loop bypass valve was successfully repaired by 1640 hours on January 20, 2001. Subsequent leak rate testing verified leakage to be 0.0839 gpm. Following successful testing of the C reactor coolant loop bypass valve the unit was brought back on line at 0254 hours on January 21, 2001. This event is reportable per 10 CFR 50.73 (a)(2)(i)(A) for completion of a nuclear plant shutdown required by the plant's Technical Specifications. No significant safety consequences resulted from this event since the leakage was contained and there was no release of radioactive material.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
NORTH ANNA POWER STATION UNIT 2	05000339	2001	- 001 -	00	2 OF 4

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

1. DESCRIPTION OF EVENT

On January 19, 2001, Unit 2 was in Mode 1 at 99 percent power (TAVG power coastdown). At approximately 0615 hours, a Reactor Coolant System (RCS) leak rate test was performed. The results indicated an increase in identified RCS (System – AB) leakage to .8038 gallons per minute (gpm). The abnormal procedure for increased primary plant leakage was entered. By 0840 hours identified leakage had increased to 2.6315 gpm. At 0944 hours, a containment entry was made to determine the source of the leakage. At the time of the entry identified RCS leakage had increased to 8.0154 gpm. The containment entry team determined that the **stuffing box** on the C reactor coolant loop bypass valve (2-RC-MOV-2587) (System – AB, Component – ISV) was the source of the leak as indicated by increased temperatures on the **packing leak off line**. By 1100 hours, identified leakage was greater than the Technical Specification (TS) limit of 10 gpm, **actual leakage was 10.0015 gpm,** and applicable actions of TS 3.4.6.2 were initiated.

At approximately 1145 hours on January 19, 2001, operators began to ramp Unit 2 offline. At this time a Notification of Unusual Event (NOUE) was declared in accordance with the EPIP 1.01, Emergency Manager Controlling Procedure, Tab B8 **Unit Shutdown Required by TS**, for exceeding the 10 gpm TS limit for RCS leakage. The actual leakage recorded at the time was 10.0015 gpm. At 1155 hours notification to state and local governments was completed. At 1215, hours a 1 hour report was made to the NRC Operations Center in accordance with 10 CFR 50.72(a)(1)(i).

On January 19, 2001, at 1735 hours, Mode 2 was entered and subsequently at 1747 hours Mode 3 was entered with the reactor shutdown. The TS action to be in hot standby (i.e., Mode 3) within six hours was exited. **The TS action to be in cold shutdown within the following 30 hours was still applicable.** A containment entry was made at 2113 hours to backseat the C reactor coolant loop bypass valve to stop the leakage. Once the valve was positioned on its backseat, at approximately 2145 hours, leakage stopped and a leak rate test was performed to confirm the identified leakage was in fact from the C reactor coolant loop bypass valve. The NOUE was terminated at 2323 hours on January 19, 2001, when RCS leakage decreased to less than 10 gpm. Identified leakage was measured at .0839 gpm. **The TS action to be in cold shutdown within the following 30 hours was exited.**

2. SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

No significant safety consequences resulted from the C reactor coolant loop bypass valve leaking past the valve stem packing material since the leakage was contained and drained to the primary drain transfer tank (PDTT) inside the containment building. The valve stem stuffing box has packing material above and below a lantern ring with the lantern ring position at the drain line to the PDTT (System – WD, Component – TK). The packing material above the lantern ring prevented leakage to the containment (System – NH) atmosphere. As such, there was no release of radioactive material. The health and safety of the public was not affected at any time during this event.

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

3. CAUSE

The increase in identified RCS leakage was the result of the C reactor coolant loop bypass valve leaking past the valve stem packing material. Each reactor coolant loop has an eight inch bypass line equipped with an isolation valve (e.g., C loop bypass valve 2-RC-MOV-2587) which is closed during normal loop operations. Each loop bypass valve is equipped with a **leak off** line to divert any valve stem leakage to the primary drain transfer tank. The cause of the stem packing material failure below the lantern ring is attributed to aging. The packing material for the C reactor coolant loop bypass valve was last replaced in 1982.

4. IMMEDIATE CORRECTIVE ACTIONS

A NOUE was declared in accordance with Emergency Plan Implementing Procedure 1.01, Emergency Manager Controlling Procedure, Tab B8, RCS leakrate requiring plant shutdown per TS 3.4.6.2. State and federal notifications were made. Unit 2 was placed in hot standby (Mode 3) in accordance with TS. Containment entry confirmed the C reactor coolant loop bypass valve as the source of the leakage. The C loop bypass valve was placed on its backseat and the leakage was isolated. All applicable TS actions were entered and exited as required.

5. ADDITIONAL CORRECTIVE ACTIONS

The C reactor coolant loop bypass valve stem packing material was replaced. Subsequent leak rate testing verified leakage to be 0.0839 gpm. Both the A and B reactor coolant loop bypass valves were inspected to ensure there was no active leakage. As a precautionary measure, the packing material for the A and B reactor coolant loop bypass valves is expected to be replaced during the next scheduled refueling outage beginning in March 2001.

The following conditions were experienced during Unit 2 shutdown and subsequent startup.

During source range channel functional testing the pre-amp test circuit response for Source Range Nuclear Instrumentation Detector, N-31, was degraded. The TS action was entered to restore the channel to operable status within 48 hours or open reactor trip breakers within the next hour. Following repairs and testing, N-31 was declared operable and returned to service.

During start-up, prior to entering Mode 2, functional testing of the auto stop oil pressure switch was performed. While making adjustments to the pressure switch, the Solid State Protection System (SSPS) train B Input Bay “1” 120 VAC supply was lost due to a blown fuse. As a result of the blown fuse, the SSPS inputs to the Loop Stop Valves, auto stop oil, reactor coolant pump bus undervoltage, reactor coolant pump bus under frequency, and

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

5. ADDITIONAL CORRECTIVE ACTIONS (continued)

reactor coolant pump breaker auxiliary contact were affected creating a train disagreement. Due to the station being in hot shutdown (mode 3) these signals were blocked by P-7 and P-8 permissives. The fuse was replaced and the testing was completed satisfactory.

At 92 percent power while ramping the unit online the individual rod position indication for Rod B8 - D bank was not moving commensurate with the other indicators for the D bank as the rods were moved in the outward directions. An incore trace confirmed that Rod B8 was in fact moving and the problem was with indication only. A signal conditioning card was replaced and Rod B8 indication returned to normal.

6. ACTIONS TO PREVENT RECURRENCE

An engineering evaluation is currently in progress. Corrective actions identified by the evaluation will be implemented as required. Historically, leakage past the valve stem packing material for the A, B, and C reactor coolant loop bypass valves is rare. Work history associated with these valves indicate they were last repacked as follows: 2-RC-MOV-2585 in 1986, 2-RC-MOV-2586 in 1982, and 2-RC-MOV-2587 in 1982.

7. SIMILAR EVENTS

Unit 1 LER, N1-1991-011-00, documents a Unit 1 shutdown due to a reactor coolant system leak from a three quarter inch upper disc pressurization line for the B cold leg loop stop valve. Report date May 11, 1991.

Unit 2 LER, N2-1991-011-00, documents a Unit 2 shutdown due to a reactor coolant system leak from the RHR inlet isolation valve stem packing material. Report date November 3, 1991.

8. ADDITIONAL INFORMATION

Unit 1 was operating in Mode 1, at 100 percent power, and was not affected by this event.

C reactor coolant loop bypass valve component information:

Mark Number	2-RC-MOV-2587
Manufacturer	Rockwell – Edwards
Model Number	7517(CF8M)JMY
Description	8 inch motor operated isolation valve