10CFR50.73

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

July 30, 1999

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

Serial No.: 99-386
SPS: BCB
Docket No.: 50-281
License No.: DPR-37

Dear Sirs:

Pursuant to 10 CFR 50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 2.

Report No. 50-281/1999-003-00

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

Enclosure

IE22

Commitments contained in this letter:

1. The Unit 1 loop stop valves will be stroked during the next refueling outage. This activity will be observed and stroke-time and stem measurement data will be obtained to confirm that the valves' stem and disc assembly are engaged.

2. Approved RCE recommendations, designed to prevent the recurrence of a similar event, will be tracked and implemented through the corrective action program.

3. Procedure changes were made to more clearly communicate the TS requirements for refilling the CST. The application of these procedures, with respect to this event, will be discussed in licensed operator requalification training.

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

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

NRC FORM 366 (6-1998) S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3 04 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)

Auto Reactor Trip on Low Coolant Flow Due to Loop Stop Valve Failure

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	DOCKET NUMBER 05000 --
07	05	1999	1999 –	003 --	00	07	30	1999	FACILITY NAME	DOCKET 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)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
E. S. Grecheck, Site Vice President	(757) 365-2000

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
D	AB	ISV	Anchor/Darling Valve Co.	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 July 5, 1999, at 11:22, the Unit 2 reactor automatically tripped from 100% power as a result of low flow in the "A" loop of the reactor coolant system (RCS). The plant responded to the reactor trip, as designed. The auxiliary feedwater pumps were secured and withdrawal from the condensate storage tank (CST) was terminated at 12:07. The Technical Specification (TS) two hour action statement for refilling the tank was inadvertently not entered, and refilling of the CST began at 19:05. At 18:10, the "A" cold leg loop stop valve, 2-RC-MOV-2591, was removed from its backseat as a part of the normal unit cooldown sequence. When an attempt was made to reopen the valve fully, a full open position was not indicated. Since compliance with TS 3.17.1 was not maintained, a 30 hour action statement to place the unit in a cold shutdown condition was entered in accordance with TS 3.0.1. A root cause evaluation (RCE) investigation determined that the low flow condition in the "A" RCS loop resulted from the separation of the 2-RC-MOV-2591 valve's disc assembly from the stem, which allowed the disc assembly to drop into the flow stream. Approved RCE recommendations, designed to prevent the recurrence of a similar event, will be implemented through the corrective action program. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv) and 10 CFR 50.73 (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 July 5, 1999, at 11:22, control room annunciators [EIIS-IB] alarmed, indicating low flow in the "A" reactor coolant loop. As a result of the low flow condition, the Unit 2 reactor automatically tripped [EIIS-JC] from 100% power and was followed by an automatic turbine trip. Control room operators promptly initiated the appropriate emergency operating procedures.

The plant responded to the reactor/turbine trip, as designed: the auxiliary feedwater pumps (AFW) [EIIS-BA-P] started on low-low steam generator (SG) water level and provided flow to the SGs; the Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) armed and initiated; and the main steam dump valves [EIIS-SB,TCV] automatically opened to admit steam to the main condenser. The temperature of the reactor coolant system (RCS) [EIIS-AB] initially decreased to approximately 542°F and subsequently stabilized at approximately 547°F. The reactivity shutdown margin was calculated to ensure that Technical Specification (TS) and administrative shutdown margin limits were satisfied.

The AFW pumps were secured and withdrawal from the condensate storage tank (CST) [EIIS-KA,TK] was terminated at 12:07. TS 3.6.H requires refilling of the CST to begin within two hours following the cessation of tank consumption. The two hour action statement was inadvertently not entered, however, and refilling of the CST began at 19:05.

The following discrepancies were noted during the post-trip response:

- The individual rod position indication (IRPI) rod bottom lights [EIIS-AA,ZI] for control rods F-6 and L-5 illuminated within one minute after the reactor trip. The IRPI rod bottom lights for F-8, G-9, and J-13 illuminated within the following 15 minutes.
- Control room annunciators alarmed approximately four minutes following the reactor trip, indicating the "A" reactor coolant pump (RCP) [EIIS-AB,P] shaft vibrations were in an alert condition. The pump was secured as a precaution.

At 18:10, the RCS "A" cold leg loop stop valve, 2-RC-MOV-2591, [EIIS-AB,ISV] was removed from its backseat as a part of the normal unit cooldown sequence. When an attempt was made to reopen the valve fully, a full open position was not indicated. TS 3.17.1 requires the loop stop valves to be maintained open unless the unit is at cold or refueling shutdown. Since compliance with TS 3.17.1 was not maintained, an action statement was entered in accordance with TS 3.0.1, requiring the unit to be placed in a cold shutdown condition within 30 hours. The unit reached cold shutdown on July 6, 1999, at 13:36, and the TS 3.0.1 action statement was exited.

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

1.0 DESCRIPTION OF THE EVENT (Continued)

The NRC was notified pursuant to 10 CFR 50.72 (b)(2)(ii) on July 5, 1999 at 13:57. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv) as an event that resulted in the automatic actuation of engineered safety features and the reactor protection system; and pursuant to 10 CFR 50.73 (a)(2)(i)(B) for operation prohibited by TS 3.6.H and TS 3.17.1.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event resulted in no safety consequences or implications. Appropriate operator actions were taken in accordance with emergency operating procedures to ensure the performance of system automatic actions. The unit was quickly brought to a stable, no-load condition. Therefore, the health and safety of the public were not affected at any time during this event.

A departure from nucleate boiling (DNB) assessment of the RCS flow transient was performed by Engineering. The assessment concluded that no DNB limits were exceeded and, therefore, no fuel failures are predicted as a result of the transient.

When the Unit 2 CST withdrawal was terminated, the tank contained approximately 88,000 gallons. Although refilling of the tank was delayed, the 60,000 gallons required to be available for Unit 1 was maintained, with ample margin.

3.0 CAUSE

A Category 1 Root Cause Evaluation (RCE) team was assembled on July 6, 1999, to determine the cause of this event and to recommend corrective actions. The RCE investigation determined that the low flow condition in the "A" RCS loop resulted from the separation of the 2-RC-MOV-2591 valve's disc assembly from the stem, which allowed the disc assembly to drop into the flow stream. An inspection of the valve internals revealed that the pin, which prevents disengagement of the stem/disc assembly threaded connection, had been sheared. The failure of the pin allowed the stem threads to almost completely disengage from the disc assembly when the valve was subsequently opened. The RCE has preliminarily concluded that the pin failure was caused by the application of excessive torque when closing the valve using the manual operator.

The CST was not refilled in accordance with TS 3.6.H as a result of the control room staff's inadvertent failure to perform the procedure that is used to respond to a CST low level alarm.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Following the reactor trip, control room operators acted promptly to place the unit in a safe, shutdown condition in accordance with emergency and other operating procedures.

The Shift Technical Advisor monitored the critical safety function status trees to ensure that plant parameters remained satisfactory.

5.0 ADDITIONAL CORRECTIVE ACTIONS

To minimize radiation dose and the amount of time the unit was at reduced inventory condition with fuel in the reactor vessel, the disc assembly was removed from 2-RC-MOV-2591. Unit operation with the disc removed was evaluated and determined to be acceptable since the loop stop valves are used for maintenance purposes and are only closed when the unit is shutdown.

Hot rod drop testing was conducted which verified that control rods F-6, F-8, G-9, J-13, and L-5 were fully operable. The respective fuel assemblies' burn-up was also evaluated and determined to be less than the burn-up threshold for incomplete rod insertion. A slow response from IRPI rod bottom indication has been exhibited following reactor trips for several years. As a result, IRPI system performance is currently being monitored relative to the condition monitoring criteria established through the Maintenance Rule program. In addition, long-term plans for system improvements have been developed.

The shaft vibration alert condition experienced by the "A" RCP following the reactor trip was evaluated. The evaluation concluded that the vibrations were an expected response to the RCS "A" loop low flow condition. A review of other RCP parameters confirmed that there were no additional problems with the pump.

6.0 ACTIONS TO PREVENT RECURRENCE

The other Unit 2 loop stop valves were stroked. While being stroked, the valves were observed and stroke-time and stem measurement data was obtained. These checks indicated that the valves' stem and disc assembly were engaged and not likely to experience a similar failure.

Based on an assessment of recent closure deflection data, the Unit 1 loop stop valves were determined to be less likely to experience a failure similar to 2-RC-MOV-2591. The Unit 1 loop stop valves will be stroked during the next refueling outage. This activity will be observed and stroke-time and stem measurement data will be obtained to confirm that the valves' stem and disc assembly are engaged.

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6.0 ACTIONS TO PREVENT RECURRENCE (Continued)

Approved RCE recommendations, designed to prevent the recurrence of a similar event, will be tracked and implemented through the corrective action program.

Procedure changes were made to more clearly communicate the TS requirements for refilling the CST. The application of these procedures, with respect to this event, will be discussed in licensed operator requalification training.

7.0 SIMILAR EVENTS

- LER No. 50-280/1987-011-01
  Reactor Trip on low RCS Flow Due to Failure of Loop Stop Valve
- December 1, 1973
  Failure of Unit 1 Loop Stop Valve

8.0 MANUFACTURER/MODEL NUMBER

Anchor/Darling Valve Company
Double-Disc Gate Valve
S350-W-DD

9.0 ADDITIONAL INFORMATION

Unit 2 was returned to service on July 16, 1999.

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