10CFR50.73

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

April 1, 2011

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

Serial No.: 11- 112
SPS: JSA
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. 50-281/2011-001-00

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

Very truly yours,

Gerald T. Bischof,
Site Vice President
Surry Power Station

Enclosure

Commitment contained in this letter: None

IE22
NRR

cc: U.S. Nuclear Regulatory Commission, Region II
Marquis One Tower, Suite 1200
245 Peachtree Center Ave., NE
Atlanta, GA 30303-1257

NRC Senior Resident Inspector
Surry Power Station

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY IMB: NO. 3150-0104 EXPIRES 08/31/2010
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose 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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Auto Reactor Trip on Low Coolant Flow Due to Loop Stop Valve Failure

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
02	02	11	2011 -	001 -	00	04	01	11	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: N

10. POWER LEVEL: 98.3%

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

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

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
B. L. Stanley, Director Safety and Licensing	(757) 365-2003

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	AB	ISV	A391	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

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

On February 2, 2011, at 0533, the Unit 2 reactor automatically tripped from 98.3% power as a result of low flow in the "C" loop of the reactor coolant system (RCS). The plant responded to the reactor trip as designed. All three auxiliary feedwater (AFW) pumps automatically initiated on low-low steam generator (SG) water level providing flow to the SGs. The AFW pumps were secured by 0613. At 1222, the reactor coolant pump for "C" loop was stopped. With no flow, compliance with Technical Specification (TS) 3.17.1 was not maintained and a 30 hour action statement to place the unit in a cold shutdown condition was entered in accordance with TS 3.0.1. A preliminary root cause evaluation (RCE) determined that the low flow condition in the "C" RCS loop resulted from a separation of a loop stop 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 10CFR50.73 (a)(2)(iv)(A) as an event that resulted in the automatic actuation of engineered safety features and the reactor protection system and pursuant to 10CFR50.73 (a)(2)(i)(B) for operation prohibited by TS 3.17.1.

NARRATIVE

1.0 DESCRIPTION OF THE EVENT

On February 2, 2011, at 0533, control room annunciator [EIIS-IB] 2E-B10, "Loss of Coolant Flow > P-8" alarmed, indicating low flow in the "C" reactor coolant loop. All three reactor coolant system (RCS) loop flow indicators [EIIS-AB, FI] for the "C" RCS loop had decreased to approximately 30% indication. As a result of the low flow condition, the Unit 2 reactor [EIIS-JC] automatically tripped from 98.3% 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: all three auxiliary feedwater (AFW) pumps [EIIS-BA, P] automatically initiated on low-low steam generator (SG) water level providing flow to the SGs and the Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) armed and initiated.

The RCS temperature decreased to approximately 539°F due to the main turbine cylinder heating steam control valves being manually bypassed in conjunction with full AFW flow. The RCS cooldown resulted in a decrease in Pressurizer level. RCS letdown isolated and Pressurizer heaters tripped as designed due to low Pressurizer level. The cylinder heating steam was isolated and AFW was reduced to mitigate RCS cooldown. Operators increased charging flow to restore Pressurizer level and letdown was restored at 0555. The AFW pumps were secured by 0613. The RCS temperature subsequently stabilized at approximately 548°F. Shutdown margin was calculated to ensure that Technical Specification (TS) and administrative shutdown margin limits were satisfied. The NRC was notified pursuant to a four hour notification of 10 CFR 50.72 (b)(2)(iv)(B) and eight hour notification of 10 CFR 50.72 (b)(3)(iv)(A).

The Source Range Nuclear Instrument (SRNI) detectors failed to energize automatically due to undercompensation of Intermediate Range Nuclear Instrument N-36 [EIIS-IL, RI]. The SRNIs were energized manually.

At 1222, the reactor coolant pump (RCP) [EIIS-AB, P] for "C" loop was stopped and no flow was indicated in "C" loop. With no flow in RCS loop "C", compliance with TS 3.17.1 requiring loop stop valves to be open unless the reactor is in cold shutdown or refueling shutdown was not maintained. Therefore, 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 February 3, 2011 at 0144, and the TS 3.0.1 action statement was exited.

This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv)(A) as an event that resulted in the automatic actuation of engineered safety features and the reactor protection system and pursuant to 10CFR50.73 (a)(2)(i)(B) for operation prohibited by 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 and the unit was quickly brought to a stable condition. Therefore, the health and safety of the public were not affected at any time during this event.

3.0 CAUSE

A Root Cause Evaluation (RCE) team was assembled to determine the cause of this event and to recommend corrective actions. The preliminary RCE determined that the low flow condition in the "C" RCS loop resulted from the separation of the 2-RC-MOV-2595 cold leg 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 the wedge pin had been sheared and the upper wedge female threads had wear resulting in disengagement of the stem to upper wedge. The preliminary RCE determined the connection of the valve stem and upper wedge of the valve internals may be susceptible to failure in a high vibration environment created from the flow turbulence of the RCP discharge when not adequately torqued after the stem and upper wedge of the valve internals are assembled. The RCE will be finalized during the upcoming refueling outage with further evaluations of old parts, additional valve diagnosis and vibration analysis.

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 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

The other Unit 2 loop stop valves were stroked and data was obtained. The troubleshooting results did not conclude degradation on the other valves.

The internals were removed from 2-RC-MOV-2595 (new disc assembly was not available) and a new stem was installed.

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6.0 ACTIONS TO PREVENT RECURRENCE

The RCE will be finalized.

Based on the preliminary RCE, cold leg loop stop valves will undergo inspection, be properly assembled including torquing the stem into the wedge and hardening of the connection of the loop stop valve stem to the upper wedge to prevent loosening of the connection. Hot leg loop stop valves will be evaluated to determine if corrective actions are required.

7.0 SIMILAR EVENTS

LER No. 50-281/1999-003-00
Auto Reactor Trip on Low Coolant Flow Due to Loop Stop Valve Failure

A similar event occurred in 1999 and the RCE team concluded inadequate lubrication of the valve stem was resulting in excessive torque being applied, breaking the anti-rotation pin and allowing the stem to unscrew from the upper wedge.

The preliminary RCE evaluated the 1999 RCE and concluded that while stem lubrication would reduce excessive torque, the actual cause of the 1999 event may be more similar to the current RCE.

8.0 MANUFACTURER/MODEL NUMBER

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

9.0 ADDITIONAL INFORMATION

Unit 1 was at 100% power and remained unaffected by the Unit 2 reactor trip.

Although there have been no failures of Unit 1 loop stop valves in this manner, they will be evaluated to determine if corrective actions are required.