10CFR50.73

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

August 3, 2010

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

Serial No.:	10-443
SPS:	JSA
Docket No.:	50-280
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 1.

Report No. 50-280/2010-003-00

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
245 Peachtree Center Ave., NE, Suite 1200
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
Surry Power Station	05000 - 280	1 OF 6

4. TITLE
Loss of Vital Bus Due to Human Error Results in Automatic Reactor Trip

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
06	08	10	2010 -	003 -	00	08	03	10	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
N	☐ 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)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☒ 50.73(a)(2)(x)
100%	☐ 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 Specify in Abstract below or in NRC Form 366A
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

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
A	EE	INVT	W120	Y	B	IG	CAP	W120	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 June 8, 2010 at 0948 hours, with Units 1 and 2 at 100% power, the Unit 1 120 VAC Vital Bus 1-III was lost during maintenance. Main Feedwater Recirculation Valve A opened and the Main Feedwater Regulating Valves re-aligned. The resultant feedwater flow/steam flow mismatch coincident with low steam generator (SG) water level on SG A, resulted in a reactor trip. Safety Injection initiated due to loss of Vital Bus instrumentation and a momentary RCS cool down. The reactor was placed in hot shutdown and the health and safety of the public were not affected. The cause was inadvertent contact of a lead during Uninterruptible Power Supply (UPS) maintenance due to human error. Procedures and training will be revised to prevent recurrence. This report is being submitted pursuant to 10CFR50.73(a)(2)(iv)(A), automatic actuation of the reactor protection system. Following the trip, a resistor-capacitor suppressor failed in the Control Room Nuclear Instrument cabinet, resulting in a small fire. The fire was quickly extinguished. Therefore, this report is also being submitted pursuant to 10CFR50.73(a)(2)(x).

NARRATIVE

1.0 DESCRIPTION OF THE EVENT

On June 8, 2010 at 0948 hours, with Units 1 and 2 at 100% power, Unit 1 120 VAC Vital Bus 1-III [EIIS-EF-BU] was lost when the Uninterruptible Power Supply (UPS) [EIIS-EF-UJX] VB 1A-2 static switch swapped from the inverter to the Regulating Line Conditioner (RLC) [EIIS-EF-90] (the alternate AC source), which was tagged out for on-going maintenance. The static switch from the inverter [EIIS-EF-ASU] swapped to the RLC because a worker dropped a lead during the maintenance. As a result, the Main Feedwater Pump Recirculation Valve A [EIIS-SJ-XCV] opened and Main Feedwater Regulating Valves (MFRV) [EIIS-SJ-FCV] B and C aligned to automatic-hold mode of operation. MFRV A, which also should have aligned to automatic-hold mode, closed due to an equipment malfunction. With this reduction in main feedwater flow, a Unit 1 automatic reactor trip occurred at 0949 hours due to a feed flow/steam flow mismatch in conjunction with low steam generator (SG) [EIIS-BA-SG] level.

Both Motor Driven Auxiliary Feedwater Pumps [EIIS-BA-P] and the Turbine Driven Auxiliary Feedwater Pump automatically started, as expected.

At 0950 hours, Safety Injection (SI) [EIIS-BQ] initiated due to high steam flow in conjunction with the low Reactor Coolant System (RCS) [EIIS-BA] average temperature. The false indication of high steam flow signal, caused by the loss of Vital Bus 1-III instrumentation, satisfied the required logic for high steam flow for all three steam lines. The loss of the Vital Bus also caused the loss of one channel of RCS temperature input into the SI logic. An additional low RCS temperature signal completed the logic for SI. During the operation of the main steam dump valves, the RCS momentarily cooled below the average no load temperature of 547°F and the SI setpoint for RCS temperature (543°F), completing the coincidence for the high steam flow with low RCS average temperature automatic SI logic. The high steam flow with RCS T_{ave} matrix also closed the Main Steam Trip Valves [EIIS-SB-TCV] and opened the Refueling Water Storage Tank [EIIS-BE-TK] cross-tie valves. The SI actuation resulted in automatic start of Emergency Diesel Generators (EDG) [EIIS-EK-DG] #1 and #3; however, the EDGs were not required since off-site power remained operable. The Component Cooling return trip valve [EIIS-CC-ISV] for Reactor Cooling Pump (RCP) 1A also closed due to the loss of the Vital Bus and RCP 1A [EIIS-AB-P] was procedurally secured.

In addition, numerous field inputs to the Plant Computer System (PCS) [EIIS-ID] were lost and resulted in non-functionality of the Safety Parameter Display System (SPDS) [EIIS-ID]. The PCS, Main Control Room (MCR) annunciators [EIIS-NA-ANN], and sufficient MCR instrumentation remained functional to monitor critical safety functions.

At approximately 1003 hours, the RCS cooled to a minimum temperature due to the combination AFW flow and SI flow. AFW flow was reduced and SI was terminated. At 1025 hours, the RCS recovered to normal temperatures with temperature control established using the SG Power Operated Relief Valves (PORVs) [EIIS-BA-SG,RV].

At 1002 hours, the Pressurizer [EIIS-AB-PZR] indicated 100% full and was confirmed solid.. The Pressurizer PORV [EIIS-AB-PZR,RV] PCV-1455C cycled to limit RCS pressure. During this time, Pressurizer Relief Tank [EIIS-AB-PZR,TANK] level increased from approximately 83% to 87% (~350 gallon increase). Charging pump [EIIS-AB-P] 1B was secured and high head SI flow to the RCS cold legs was isolated. At 1016 hours, Chemical and Volume Control System [EIIS-CB] Letdown flow was restored. At 1034 hours, a Pressurizer bubble was re-established. Unit 1 was stabilized at Hot Shutdown.

At 1200 hours, a non-emergency, four-hour and eight-hour notification was made to the NRC pursuant to 10CFR 50.72(b)(2)(iv)(B), any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical, 10CFR 50.72(b)(2)(iv)(A) for ECCS discharge to RCS, 10CFR 50.72(b)(3)(iv)(A) for specified system actuations of the EDG and AFW, and 10CFR 50.72(b)(3)(xiii) for loss of the SPDS portion of the PCS.

This report is being submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A), actuation of the reactor protection system (RPS).

Subsequent to the trip, at 1121 hours, a resistor-capacitor (RC) suppressor in the Unit 1 Channel II Power Range Nuclear Instrumentation (PRNI) [EIIS-IG] cabinet, located in the Main Control Room, failed and the resin encapsulating the RC suppressor caught fire. An RC suppressor located approximately three inches above also caught fire. The small fire was quickly extinguished at 1124 hours with a hand held fire extinguisher. The RC suppressor failure was not directly associated with the loss of the Unit 1 Vital Bus III, but was attributed to the normally de-energized RC suppressor which energized as a result of the trip. This RC suppressor is not energized when the unit is greater than 23% power but energizes below this point. Due to the location of the fire, this report is also being submitted pursuant to 10CFR50.73(a)(2)(x).

After the initial RC suppressor failure, an odor was detected in a different Unit 1 PRNI cabinet and a second failed RC suppressor was discovered. There was no indication of fire or smoke and the event terminated when the fuse opened.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

Operations response and procedure usage was appropriate. The operators properly diagnosed the imminent nature of an SI signal upon loss of the Vital Bus and transitioned through abnormal and emergency procedures appropriately. An initial evaluation determined that although this event is considered complicated, with the initiation of the SI, cycling of the Reactor Coolant System Power Operated Relief Valves, and partial loss of feedwater, the Conditional Core Damage Probability was determined to be low.

The small fire was extinguished quickly, did not significantly hamper the operators, nor threaten plant equipment. An initial evaluation concluded that given limited potential damaged targets due to proximity of the affected electrical cabinets, the Conditional Core Damage Probability due to the fire event was low.

No Emergency Plan entry criterion was met for either event. There were no radiation releases due to these events. Therefore, the health and safety of the public were not affected.

3.0 CAUSE

The direct cause for the reactor trip was a feed flow/steam flow mismatch in coincident with low S/G level on the S/G A. The loss of feedwater flow to the S/G A occurred when Vital Bus 1-III was lost during maintenance activities on the 1A2 UPS RLC. The vital bus de-energized when a lead that was being re-landed on the RLC Low Voltage card contacted an adjacent component. The inadvertent contact caused the static switch to swap from the inverter to the RLC which was tagged out for maintenance.

The root cause was identified as human error when workers dropped a lead during performance of maintenance on the UPS. A knowledge deficiency on the operation of the UPS resulted in a lack of understanding of the potential consequences of the evolution. Also, the procedure that implements operational risk assessments did not require screening for this preventive maintenance activity. This activity has been successfully performed 11 times on the station's UPSs prior to this event.

The most probable cause of the RC suppressor failure was hardening and cracking of the epoxy insulation due to aging resulting in circuit failure inside the suppressor. These assemblies are original plant equipment and the failure of the capacitor allowed sufficient current to flow through the resistor, resulting in overheating the surrounding epoxy to the point of ignition.

Causal evaluations were initiated for the closure of the MFRV A and the RCS temperature decrease below the average no load temperature of 547°F. Results of the cause evaluations will be documented in the corrective action program.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

When Unit 1 tripped due to loss of Vital AC Bus 1-III, operators acted promptly to place the plant in a safe, hot shutdown condition in accordance with emergency and other operating procedures.

The small fire was extinguished quickly and the affected NI channel remained energized with no additional RPS actuations.

Interim actions were implemented to require additional management review for maintenance associated with energized leads or around energized equipment.

5.0 ADDITIONAL CORRECTIVE ACTIONS

Causal evaluations were initiated to investigate the cause of the reactor trip and the RC suppressor failure. RC suppressors in the Unit 1 NI cabinets were replaced prior to exceeding hot shutdown. RC suppressors in the Unit 2 NI cabinets have also been replaced.

Corrective actions from the cause evaluations for closure of the MFRV A and the RCS temperature decrease will be implemented in accordance with the corrective action program.

6.0 ACTIONS TO PREVENT RECURRENCE

Procedures will be revised to address the performance of operational risk assessment for routine maintenance and to add cautions on grounding consequences. Training programs will be revised to address the knowledge gap, review this event in training sessions, and to address techniques for lifting and landing leads.

Non-fire retardant RC suppressors in similar applications will be replaced with qualified replacements. Preventive maintenance for RC suppressor will be implemented.

7.0 SIMILAR EVENTS

There were no similar events associated with a reactor trip following a loss of the Vital Bus. The specific maintenance has been successfully performed many times prior to this event.

A similar RC suppressor failure occurred during Unit 2 reactor start-up following the

2009 Fall Refueling Outage. The condition report that documented the failure lacked sufficient detail. As a result, appropriate evaluations were not performed to address causes and extent of condition.

8.0 MANUFACTURER/MODEL NUMBER

Westinghouse Electric Corporation, Component Part Number: PMR-2026/0.25+150

9.0 ADDITIONAL INFORMATION

Unit 2 was at 100% power and remained unaffected by the Unit 1 reactor trip. RC suppressors in the Unit 2 NI cabinets were also replaced.