

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

March 26, 2003

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

Serial No.: 03-198
SPS: BAG/TJN R2
Docket No.: 50-280
License No.: DPR-32

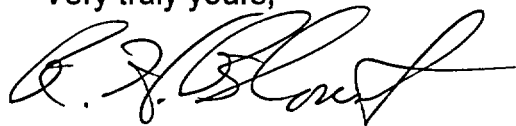
Dear Sirs:

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. 50-280/2003-002-00

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,



Richard H. Blount,
Site Vice President
Surry Power Station

Enclosure

Commitment contained in this letter:

Two Root Cause Evaluation (RCEs) were initiated to determine the causes of these events. The approved recommendations from the RCEs necessary to prevent recurrence will be implemented through the corrective action program.

IE22

cc: United States Nuclear Regulatory Commission

Region II
Sam Nunn 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

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 hours Reported lessons learned are incorporated into the licensing process and fed back to industry Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@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 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) SURRY POWER STATION , Unit 1	DOCKET NUMBER (2) 05000 - 280	PAGE (3) 1 OF 4
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TITLE (4)
Manual Steam Generator Level Control Results in Power Ascension Reactor Trip

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
01	25	2003	2003	-- 002 --	00	03	26	2003	FACILITY NAME	DOCUMENT NUMBER 05000-
									FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9) N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)									
POWER LEVEL (10) 27 %		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)		X 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		
		20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A		
		20.2203(a)(2)(iv)		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 Richard H. Blount, Site Vice President	TELEPHONE NUMBER (Include Area Code) (757) 365-2000
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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	BA	65	W290	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

<input checked="" type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE)	<input type="checkbox"/> NO	EXPECTED SUBMISSION DATE (15)	MONTH 09	DAY 30	YEAR 2003
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ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 25, 2003, at 0700 hours a Unit 1 automatic reactor trip occurred during startup at 27% reactor power due to low-low level in "B" steam generator. Automatic actuations occurred as expected, including Turbine Trip by Reactor Trip, and Main Generator Trip. The Turbine Driven Auxiliary Feedwater Pump (TDAFWP) started automatically, but tripped on overspeed. The two Motor Driven Auxiliary Feedwater Pumps automatically started and provided flow. In addition, the cross-connect from the Unit 2 AFW system remained operable. The operating staff acted promptly and appropriately to stabilize the unit at hot shutdown. A root cause evaluation (RCE) was initiated and the preliminary cause is difficult manual control of steam generator water levels at low power due to a modification made to the main feedwater regulating valves. A RCE was also initiated for the overspeed trip of the TDAFWP. The TDAFWP governor was replaced, and post maintenance testing completed on January 28, 2003. Training was provided to the operating shift responsible for the unit re-start. Unit 1 was restarted and achieved 100% reactor power on January 31, 2003 at 0755 hours. This report is being submitted in accordance with 10CFR50.73(a)(2)(iv)(A).

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		YEAR 2003	SEQUENTIAL NUMBER -- 002 --	REVISION NUMBER 00	

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

1.0 DESCRIPTION OF THE EVENT

On January 25, 2003 at 0619 hours, Unit 1 was placed online following an eleven day forced outage due to high vibrations on a reactor coolant pump motor (reference LER 50-280/2003-001-00).

During power ascension with the main feedwater regulating valves (MFRVs) in manual mode of operation, the level on 'B' steam generator (SG) was observed to be decreasing. In response, the operating crew attempted to maintain level by isolating the 'B' SG Blowdown. At 0700 hours, an automatic reactor trip occurred at 27% reactor power due to a low-low level in 'B' SG. Automatic actuations occurred as expected, including Turbine Trip by Reactor Trip, and Main Generator Trip. Both Motor Driven Auxiliary Feedwater Pumps (MDAFWPs) automatically started due to the SG low-low level. At approximately 0701 hours, the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) started, however, the pump tripped on overspeed approximately 1 minute later and was declared inoperable. The two MDAFWPs continued to run to provide the necessary heat sink. The combination of Steam Dump Valves remaining open, relatively low decay heat, inflow of cold AFW, and leakage past the Cylinder Heating Steam Valves, allowed the reactor coolant system (RCS) to cool down to an average temperature (Tave) of 543 degrees Fahrenheit (F) where the Steam Dump Valves automatically closed as designed. The RCS cool down continued to approximately 522 degrees F. In accordance with emergency operating procedures, the Main Steam Trip Valves were closed, AFW flow was throttled closed, and SG Power Operated Relief Valves were set to stabilize the RCS Tave to a no load value of 547 degrees F.

At 0925 hours on January 25, 2003, a four-hour and eight-hour non-emergency report was made to the NRC as required by 10 CFR 50.72(b)(2)(iv)(B) and 10 CFR 50.72 (b)(3)(iv)(A), respectively. This report is being submitted pursuant to 10 CFR 50.73(a) (2)(iv)(A) for an automatic actuation of the reactor protection system (RPS) and the initiation of the AFW system.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event resulted in no significant safety consequences or implications. Although the TDAFWP tripped on overspeed after automatically starting, the two MDAFWPs automatically started and continued to provide the necessary heat sink. In addition, the cross-connect from the Unit 2 AFW system remained operable. All other emergency systems functioned as required for the reactor trip. Prior operability of the TDAFWP was demonstrated on January 14, 2003 when Unit 1 was manually tripped and the TDAFWP pump automatically started and provided auxiliary feedwater to SGs (reference LER 50-280/2003-001-00).

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

The shutdown margin for Unit 1 was determined to be satisfactory. All electrical busses transferred properly following the trip and all emergency diesel generators were operable. The RCS cooled to a minimum Tave of approximately 522 degrees F and then was stabilized to the no load Tave value of 547 degrees F. The operating crew acted promptly and appropriately to stabilize the unit at hot shutdown. Therefore, the health and safety of the public were not affected.

3.0 CAUSE

In 2001, the Unit 1 MFRVs were modified in an attempt to improve low power feedwater control. However, when the unit was re-started following the modification, the MFRVs demonstrated worsened performance. This condition has been determined to be the preliminary root cause for the event described in this LER.

A RCE was also initiated to determine the cause of the TDAFWP overspeed trip. The preliminary cause is degraded governor [EISS- BA, 65] performance.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

The feedwater system engineer reviewed flow characteristics of the new style MFRV with the re-start operating crew. In addition, the training simulator was modified to replicate the response characteristics of the new style MFRVs and the operating crew responsible for the re-start of the unit received start-up simulator training. Lastly, a training synopsis was developed to assist operators on the use of feedwater flow and steam flow instrumentation at low power levels as an anticipatory means to control SG water level.

The TDAFWP governor was replaced and satisfactory post maintenance testing was completed. The TDAFWP pump was returned to operable status on January 28, 2003.

The unit was successfully restarted and achieved 100% reactor power on January 31, 2003 at 0755 hours.

5.0 ADDITIONAL CORRECTIVE ACTIONS

Modifications to improve MFRV performance are planned for the upcoming 2003 refueling outage.

6.0 ACTIONS TO PREVENT RECURRENCE

A RCE was initiated to determine the cause for the problems with controlling SG level in manual at low power. A RCE was also initiated for the TDAFWP overspeed trip.

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

Conclusions from these RCEs will be evaluated and the approved recommendations from the RCEs necessary to prevent recurrence will be implemented through the corrective action program.

7.0 SIMILAR EVENTS

LER 50-281/96-04-00, Turbine/Reactor Trip Due to High Level in the Steam Generator

With power escalation in progress for Unit 2, the operators began transferring from manual feedwater flow control to automatic control. At 16% reactor power, a high-high SG 'B' level signal caused a turbine trip and subsequent reactor trip. The root cause of the trip was design interface. Specifically, manual SG level control at low power, combined with other equipment malfunctions, challenged the operating team to the point where the SG level could not be successfully controlled. A design change was implemented to replace the feed regulating valve on Unit 1 as is noted in the above discussions.

8.0 MANUFACTURER/MODEL NUMBER

The TDAFWP governor that was replaced was a Woodward PG-PL.

9.0 ADDITIONAL INFORMATION

None