10CFR50.73

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

February 9, 2004

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

Serial No.:	03-636
SPS: BAG/TJN	R1'
Docket No.:	50-280
	50-281
License No.:	DPR-32
	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 Units 1 and 2.

Report No. 50-280, 50-281/2003-006-00

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,

R. H. Blount

Richard H. Blount,
Site Vice President
Surry Power Station

Enclosure

Commitment contained in this letter:

1. A modification to install stop check valves downstream of each of the auxiliary feedwater steam generator isolation motor operated valves, prior to where the two headers join together, will be performed during the refueling outages on Unit 1 and Unit 2.

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. G. J. McCoy
NRC Senior Resident Inspector
Surry Power Station

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104	EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
SURRY POWER STATION , Unit 1	05000 - 280	1 OF 5

TITLE (4)

Steam Generator AFW Isolation Unanalyzed Condition from Original Design

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
									Surry Power Station, Unit 2	05000-281
12	12	2003	2003	-- 006 --	00	2	9	2004	FACILITY NAME	DOCKET NUMBER 05000-

OPERATING MODE (9)	N
POWER LEVEL (10)	100 %

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

	20.2201(b)		20.2203(a)(3)(ii)	X	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
	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)	X	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)
Richard H. Blount, 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
N/A									

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 December 12, 2003, at 0900 hours with Units 1 and 2 at 100% power, it was determined that for both Unit 1 and Unit 2, an unanalyzed condition existed with the alignment configuration of the normally open auxiliary feedwater (AFW) motor operated valves (MOVs). The AFW system isolation function assumed by the accident analysis was not capable of being performed during a steam generator tube rupture event concurrent with the loss of one emergency bus.

The root cause for this condition was design configuration/ analysis in that the unanalyzed condition was not considered during initial plant design. Unit 1 and Unit 2 AFW systems were declared inoperable at 1119 hours on December 12, 2003 and a 6-hour clock to hot shutdown was entered for both Units. It was determined that maintaining four of the six AFW MOVs closed and two AFW MOVs open would return the unit to an analyzed condition. When the AFW MOVs were placed in this alignment, both Units 1 and 2 exited the 6-hour clock at 1341 hours on December 12, 2003. There were no significant safety consequences associated with this condition. Stop check valves will be installed to prevent back flow though the AFW MOVs. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(ii)(B).

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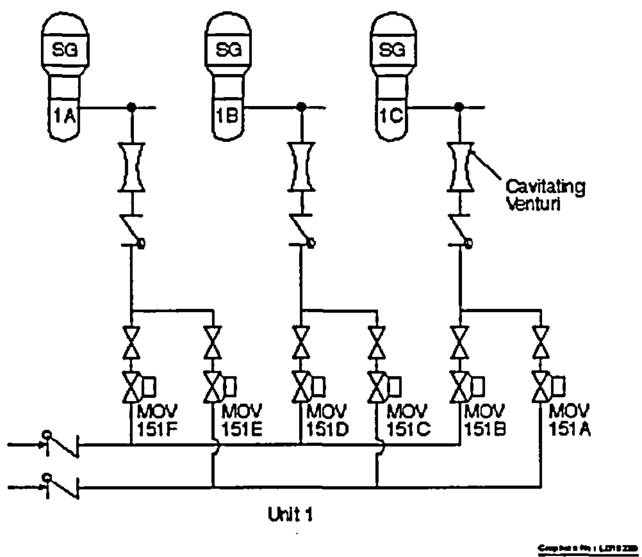
NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

1.0 DESCRIPTION OF THE EVENT

During a simulator training session it was identified that a potential problem existed with the methodology used to control auxiliary feedwater (AFW) flow to the steam generators (SGs) [EIIS- BA, SG] under accident conditions when one emergency bus is de-energized. A corrective action system plant issue was initiated with actions assigned to further review this condition.

The Surry AFW system has three AFW pumps feeding two headers, which in turn supply three steam generators. There are six motor operated valves (MOVs) inside containment, two for each SG, which are used to control flow from the two AFW headers. For each SG, one MOV is powered from the "H" emergency bus, and one MOV from the "J" emergency bus. The MOVs are normally maintained open (and receive an open signal concurrent with AFW pump start signals).



A review of this scenario determined that a ruptured steam generator condition was not bounded by the accident analyses, and therefore was an unanalyzed condition. Under steam generator tube rupture (SGTR) accident conditions with either the "H" or "J" emergency bus de-energized, the three MOVs powered from the de-energized emergency bus cannot be repositioned closed unless a containment entry is made. Therefore, only one of the two MOVs to the ruptured SG could be closed by operating the MOVs remotely while the intact SGs are being filled to meet heat sink requirements. Operation of one motor driven AFW pump and the turbine driven AFW pump will provide more flow to the intact SGs and some flow is directed towards the ruptured SG. Since there are no check valves in each header downstream of the MOV prior to the headers joining together, flow from the energized train of piping will flow back through the de-energized open MOVs into the deenergized train of piping, and into the ruptured SG.

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

On December 12, 2003 at 1119 hours, with Units 1 and 2 at 100% reactor power, a second plant issue was submitted documenting the results of the review. Unit 1 and Unit 2 AFW systems were declared inoperable and a 6-hour clock to hot shutdown (HSD) was entered for both units in accordance with Technical Specification (TS) 3.0.1.

At 1433 hours, on December 12, 2003, an eight-hour notification was made to the NRC of the unanalyzed condition pursuant to 10 CFR 50.72(b)(3)(ii)(B).

This report is provided pursuant to 10 CFR 50.73(a)(2)(i)(B), for the inoperable AFW conditions prohibited by TSs, and 10 CFR 50.73(a)(2)(ii)(B), for the plant being in an unanalyzed condition.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

The existing accident analysis for a SGTR assumes auxiliary feedwater isolation to a ruptured steam generator within 30 minutes. Isolation of AFW to a ruptured SG may not be possible within the time frame specified in the analysis. For this specific SGTR scenario, failure to isolate AFW could result in a radioactive release or failure of equipment important to safety, which has not been previously analyzed.

A probabilistic risk assessment found that the unisolable steam generator increased the risk contribution from a SGTR sequence, however, the actual risk contribution was small, and was a negligible portion of the total core damage frequency. For this specific scenario the contribution was less than 6E-9 per year.

Given these considerations, the interim measures and actions discussed in Sections 4.0 and 5.0, and the fact that a SGTR concurrent with loss of one emergency bus did not occur, this condition resulted in no significant safety consequences, and the health and safety of the public were not affected.

3.0 CAUSE

The current AFW configuration has existed since original construction. No documentation was found that indicated the AFW MOVs have ever been required to be in a special alignment to account for this condition. The root cause for this condition is design configuration/ analysis, in that the unanalyzed condition was not considered during initial plant design.

The previous simulator computer did not model backflow through the AFW MOVs. A recent upgrade to the model more accurately reflected the plant design and showed that the steam generator in the simulator training session continued to fill after the energized AFW MOVs were closed.

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

4.0 IMMEDIATE CORRECTIVE ACTION(S)

A plant issue was submitted at 1119 hours on December 12, 2003, describing that the isolation function of the AFW system is not capable of being performed as required by accident analysis under design basis SGTR conditions, which include loss of one train of emergency power. Unit 1 and Unit 2 AFW systems were declared inoperable and a 6-hour clock to hot shutdown was entered for both Units.

5.0 ADDITIONAL CORRECTIVE ACTIONS

Engineering provided technical information to return the Units to an analyzed condition by closing 4 of the 6 AFW MOVs. The Station Nuclear Safety and Operating Committee (SNSOC) approved the recommended actions and 4 of the 6 AFW MOVs were closed. With the units within the assumptions for the accident analysis, the AFW system was declared operable and the 6-hour clock to HSD was exited at 1341 hours on December 12, 2003.

A justification for continued operation (JCO) was approved to document the required compensatory measures to maintain the units within the accident analysis, and a temporary modification (TM) was implemented to restrict AFW MOV valve movement. Procedurally controlled TMs were subsequently implemented to allow MOV testing or system realignment while continuing to meet the JCO.

An extent of condition review was performed. The Surry accidents are modeled using NRC approved industry wide computer codes, and an input table is used to specify assumed values for Auxiliary Feedwater. In addition, the Surry AFW system is of a customized design (three pumps feed two headers, which supply three steam generators), and is not an industry standard. Other safety related systems at Surry are closer in design to a standard plant which has a larger amount of industry experience for analysis. Based on the above information, it is unlikely that configuration problems similar to that discovered on the AFW system exist for other Surry systems.

6.0 ACTIONS TO PREVENT RECURRENCE

A design change package will be developed to install stop check valves downstream of each of the AFW steam generator isolation MOV, prior to where the two headers join together. These modifications will be performed during the refueling outages on both Units, and will allow exit from the JCO.

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

7.0 SIMILAR EVENTS

None.

8.0 MANUFACTURER/MODEL NUMBER

Not applicable.

9.0 ADDITIONAL INFORMATION

None.