NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Surry Power Station, Unit 2	0 5 0 0 0 2 8 1	1 OF 0 3

TITLE (4)
Reactor Trip-Feedwater Isolation

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 NAMES	DOCKET NUMBER(S)
1 1	2 0	8 4	8 4	— 0 1 6 —	0 0	1 2	2 0	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 2 1

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[X] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
R. F. Saunders, Station Manager	804 357-3184

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

With reactor power at 21 percent and increasing, the reactor tripped on low water level in "C" steam generator. The low level in "C" steam generator was caused when a unit startup was commenced with the main feedwater manual isolation valves closed. This was due to both a loss of administrative control and personnel error.

Administrative controls (startup procedures) are being modified to include a verification of feedwater valve position and precautions to personnel to use Control Board mounted tagging notes. Personnel involved were also re-instructed in correct plant practices. In addition, Vepco's Human Performance Evaluation System (HPES) was utilized to investigate and make recommendations on improving personnel performance.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

1. Description of the Event

On November 20, 1984, a unit two startup was in progress after a maintenance outage with the reactor power at 21% and turbine power at 44 MWe. Feedwater control was in manual and the transition from the bypass to the main feed regulating valves (FRV) (EIIS FCV) was commenced on "C" steam generator in response to a low water level. At the same time, the operator had accelerated the turbine ramp rate in an attempt to eliminate an indicated anti-motoring turbine trip alarm. The "C" FRV indicated approximately 80% open demand when the operator noticed no feedwater flow. Upon local inspection, the manual isolation valves for main feedwater to all steam generators were discovered closed.

An attempt was made to unisolate main feedwater flow to increase steam generator water level, but a reactor trip ensured as a result of low water level in "C" steam generator at approximately 0419 hours.

During this event, all control and safety systems functioned properly. Operators followed appropriate plant procedures and stabilized the plant following the reactor trip.

2. Safety Consequences and Implications

The worst case loss of feedwater accident, as analyzed in the UFSAR, assumes a reactor power of 102% with low water level in all steam generators. This analysis does not result in any adverse condition in the core, because it does not result in water relief from the pressurizer reliefs or the safety valves, nor does it result in an uncovering of the tube sheets of the steam generators.

Prior to this reactor trip, reactor power was approximately 21%. Feedwater remained available via the main feed by-pass lines and the auxiliary feedwater system. During this event, all safety related systems remained operable and plant parameters remained within the bounds of the accident analysis. Therefore, this event did not constitute an unreviewed safety question or effect the health and safety of the public.

3. Cause

The cause of the low level in "C" steam generator was due to an increasing power level with the manual feedwater isolation valves closed. These valves were closed by direction of Periodic Test 15.3 (Main and Auxiliary Feedwater Cold Shutdown Check Valve Test) which was completed during the maintenance outage. These valves were not reopened to minimized the possibility of overfilling the steam generator.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

The manual feedwater isolation valves remained closed during startup due to:

1) Failure of operators to convey and track off normal plant status.

2) Failure to follow administrative procedures.

4. Immediate Corrective Action

An unsuccessful attempt was made to open the manual isolation valves to increase steam generator level.

5. Additional Corrective Action

Following the reactor trip, the operators performed all appropriate emergency procedures and function restoration procedures to ensure the plant was returned to a stable condition. Also, the STA performed the status tree reviews to ensure specific plant parameters were noted and appropriate procedures were used to maintain those parameters within safe bounds.

6. Action Taken to Prevent Recurrence

The following actions were initiated:

1) The plant startup procedures are being modified to include checking feedwater valve positions and stroke testing feedwater valves. Precautions are also being added to remind personnel to use control board mounted tagging notes.

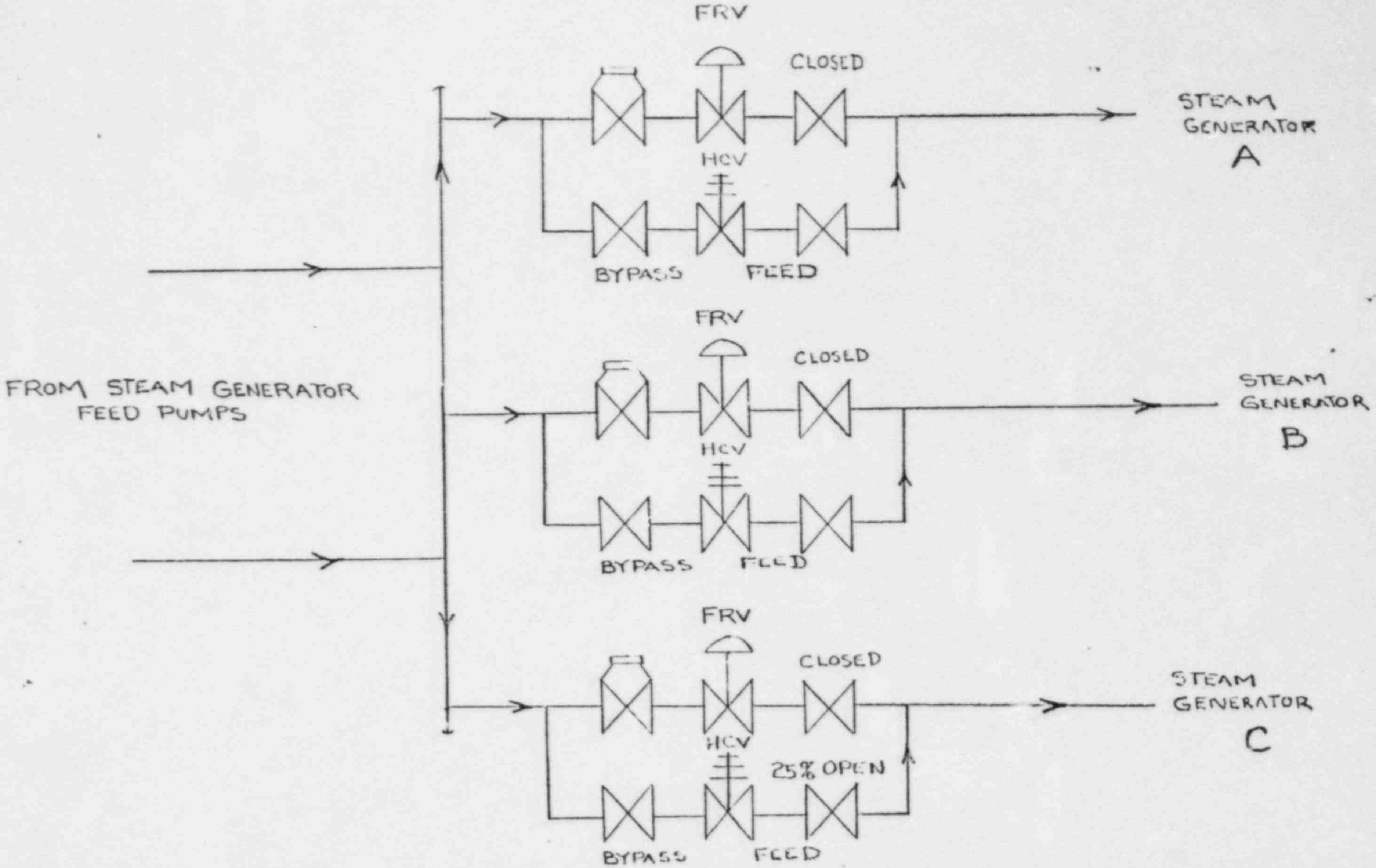
2) Personnel involved were re-instructed in correct plant practices, for maintaining administrative control of off normal plant conditions.

3) Training for plant operators will include material concerning this event.

4) Vepco's implementation of INPO's Human Performance Evaluation System (HPES) was used to investigate and to make recommendations on improving personnel performance.

7. Generic Implications

None.


FROM STEAM GENERATOR
FEED PUMPS
FRV
CLOSED
HCV
BYPASS
FEED
STEAM
GENERATOR
A
FRV
CLOSED
HCV
BYPASS
FEED
STEAM
GENERATOR
B
FRV
CLOSED
HCV
25% OPEN
BYPASS
FEED
STEAM
GENERATOR
C

Vepco

VIRGINIA ELECTRIC AND POWER COMPANY
Surry Power Station
P. O. Box 315
Surry, Virginia 23883

DEC 20 1984

Serial No: 84-041

Docket No: 50-281

License No: DPR-37

U. S. Nuclear Regulatory Commission
Document Control Desk
016 Phillips Building
Washington, D.C. 20555

Gentlemen:

Pursuant to Surry Power Station Technical Specifications, the Virginia Electric and Power Company hereby submits the following Licensee Event Report for Surry Unit 2.

REPORT NUMBER

84-016-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be reviewed by Safety Evaluation and Control.

Very truly yours,

RF Saunders
R. F. Saunders
Station Manager

Enclosure

cc: Mr. James P. O'Reilly
Regional Administrator
Suite 2900
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Atlanta, Georgia 30323

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