10CFR50.73

Virginia Electric and Power Company
Surry Power Station
P. O. Box 315
Surry, Virginia 23883

April 27, 1994

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

Serial No.: 94-280
SPS:BCB
Docket No.: 50-280
License No.: DPR-32

Dear Sirs:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 1.

REPORT NUMBER

50-280/94-005-00

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,

M. R. Kansler
Station Manager

Enclosure

cc: Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

IE22

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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)
Steady State Reactor Power Exceeded Operating License Limit

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
03	30	94	94	-- 005 --	00	04	27	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): N
POWER LEVEL (10): 100.5

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

	20.402(b)		20.405(c)		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)	X	OTHER
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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)		Voluntary

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
M. R. Kansler, 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 15 single-spaced typewritten lines) (16)

On March 30, 1994, Unit 1 completed power ascension following a refueling outage. When the unit reached 100% power, control room operators noted that the main generator gross output indication was high, up to 835 megawatts-electrical (MWe). Station management reviewed the condition and directed Operations personnel to reduce power to 820 MWe until plant parameters could be evaluated by the Engineering Department. At 2054 hours, unit power was reduced to approximately 98%. Engineering personnel investigated the discrepancy and determined that the Prodac 250 computer system FLOWCALC program, which computes main steam and feedwater flows, had not been updated to incorporate the most recent revision of the main steam flow transmitter differential pressure span values. Output from the FLOWCALC program is used by the CALCALC program which provides an indication of reactor power. Further investigation indicated that the unit had operated at a shift average of approximately 100.5% power, 2453 megawatts-thermal. This event was caused by an inadequate instrumentation change control process. To prevent recurrence, the instrumentation change control process will be modified to ensure changes to station instrumentation are effectively controlled. The event is being reported voluntarily as a violation of the facility operating license.

1.0 DESCRIPTION OF THE EVENT

On March 29, 1994, Unit 1 was completing power ascension following a refueling outage. At 1200 hours, control room operators noted discrepancies in the various indications of reactor power (i.e., power range nuclear instrumentation, Reactor Coolant System ΔT, steam flow calorimetric, turbine power) and stopped the evolution at approximately 87% reactor power. Engineering personnel reviewed the power indications and verified the inputs to the Prodac 250 (P-250) computer system [EIIS-ID] steam flow/feedwater flow calorimetric program (CALCALC). The indications of reactor power were not significantly different from those noted during the previous fuel cycle and were close to the values expected at that point in the evolution. Based on the results of the Engineering Department's review, the power range nuclear instrumentation [EIIS-IG] was adjusted to be consistent with the steam flow calorimetric and power ascension was resumed.

On March 30, 1994, at approximately 1418 hours, the unit reached 100% power as indicated by the steam flow calorimetric. Control room operators noted that the main generator [EIIS-TB] gross output indication was high, up to 835 megawatts-electrical (MWe), although within the confines of previous operational experience. The high generator output and other associated parameters were monitored for several hours. Station management reviewed the condition and directed Operations personnel to reduce power to 820 MWe until plant parameters could be evaluated by the Engineering Department. At 2054 hours, unit power was reduced to approximately 98%.

Engineering personnel investigated the discrepancy and determined that the P-250 computer system FLOWCALC program, which computes main steam and feedwater flows, had not been updated to incorporate the most recent revision of the main steam flow transmitter differential pressure span values. The main steam flow transmitters [EIIS-SB,FT] had been spanned during the refueling outage. Output from the FLOWCALC program is used by the CALCALC program which provides an indication of reactor power based on steam flow or feedwater flow.

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

1.0 DESCRIPTION OF THE EVENT (Continued)

Engineering personnel initially determined that the actual reactor power was approximately 101%, not including the measurement uncertainties, and that safety analysis limits had not been exceeded. Further investigation confirmed that the unit had operated at a shift average of approximately 100.5% power, 2453 megawatts-thermal (MWt).

The Facility Operating License, No. DPR-32, authorizes Unit 1 operation at a steady state power level not to exceed 2441 MWt. Since this steady state power level was exceeded, the event constitutes a violation of the facility operating license and is being reported voluntarily due to its potential safety significance and general interest to other licensees.

2.0 SAFETY CONSEQUENCES AND IMPLICATIONS

This event resulted in no safety consequences or implications. The Engineering Department evaluated the effect of the FLOWCALC program error on the values used in the CALCALC program, and performed calculations to determine the actual reactor power level during the event. The calculations were conservative and indicated that reactor power did not exceed 101.78%. This event is bounded by Surry's accident analyses, which includes 2.0% uncertainties. Therefore, the health and safety of the public were not affected.

3.0 CAUSE

This event was caused by an inadequate instrumentation change control process. The steam flow transmitters had been spanned during the Unit 1 refueling outage. The FLOWCALC program was not revised at that time since the implementation of the instrumentation change and required computer program change were not coordinated or verified.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Station management reviewed the main generator electrical output indication and other parameters, and directed that the unit power be reduced. Unit power was reduced to approximately 98% at 2054 hours.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

On March 31, 1994, Engineering personnel began an investigation to determine the cause of the high electrical output indication from the main generator. The investigation indicated that the FLOWCALC program had not been updated to incorporate the most recent revision of the main steam flow transmitter differential pressure span values. To immediately correct the effect of this error on reactor power indication, the power range nuclear instrumentation was adjusted and the Reactor Coolant System ΔT indications [EIIS-AB] were calibrated based on the feedwater flow calorimetric which was considered to be the most accurate indication of reactor power. These indications had previously been based on the steam flow calorimetric.

A Root Cause Evaluation team was promptly formed to investigate the cause of this event and to recommend appropriate corrective actions.

The FLOWCALC program was revised to incorporate the correct steam flow transmitter differential pressure span values.

The power range nuclear instrumentation was adjusted to the corrected steam flow calorimetric and the unit was returned to 100% power on April 5, 1994.

6.0 ACTIONS TO PREVENT RECURRENCE

The instrumentation change control process will be modified to ensure changes to station instrumentation are effectively controlled.

The Root Cause Evaluation team will complete its investigation and present recommendations to management by the end of April, 1994.

7.0 SIMILAR EVENTS

None

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

8.0 MANUFACTURER/MODEL NUMBER

N/A

9.0 ADDITIONAL INFORMATION

Unit 2 was operating at 95.5% power and was not affected by this event.

An Operating Experience Event Report will be entered into INPO's Nuclear Network computer system to notify other nuclear utilities of this event.