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AUTH.NAME AUTHOR AFFILIATION
KANSLER,M.R. Virginia Power (Virginia Electric & Power Co.)
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-014-01:on 910724,determined that scaling for main steam flow transmitters incorrect,resulting in setpoints in excess of max allowed value of 110%.Caused by use of incorrect methodology.Scaling program underway.W/920219 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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Virginia Electric and Power Company
Surry Power Station
P. O. Box 315
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February 11, 1992

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

Serial No.: 91-501A
Docket No.: 50-280
50-281
License No.: DPR-32
DPR-37

Gentlemen:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following updated Licensee Event Report for Units 1 and 2.

REPORT NUMBER

91-014-01

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be reviewed by the Corporate Management Safety Review Committee.

Very truly yours,

M. R. Kansler
Station Manager

Enclosure

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

IE22
1/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Surry Power Station, Unit 1	0 5 0 0 0 2 8 0	1 OF 0 4

TITLE (4) Main Steam Flow Setpoints in Excess of Technical Specifications Due to Flow Transmitter Scaling Inaccuracies Resulting From Incorrect Scaling Methodology

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)
0 7	2 4	9 1	9 1	0 1 4	0 1	0 2	1 1	9 2	Surry, Unit 2	0 5 0 0 0 2 8 1
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

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)	[ ]	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)	[X]	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)
M. R. Kansler, Station Manager	8 0 4 3 5 7 - 3 1 8 4

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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)

On July 24, 1991, with Unit 1 and Unit 2 operating at 100% power, it was determined after analyzing recent special test data that the scaling for the Unit 2 Main Steam (MS) flow transmitters was incorrect. These flow transmitters provide input to certain Reactor Protection System (RPS) and Engineered Safety Features (ESF) actuation signals. It was determined that the scaling resulted in Unit 2 setpoints in excess of the maximum allowed value of 110% (at full load) of full steam flow specified in Technical Specification (TS) 3.7.D. On July 22, 1991, voltage biases were placed on the uncompensated Unit 2 steam flow signals to reduce the Hi Steam Flow ESF function actuation to a value below 110%. This event was caused by the use of incorrect scaling methodology, employed in 1977, to rescale the MS flow transmitters to correct an observed difference in the MS and feedwater flow indications. Since the same incorrect scaling information was used in 1977 for Unit 1, similar voltage biases were introduced on Unit 1 on July 24, 1991. Following an evaluation of special test data, the Unit 2 steam flow and feedwater flow transmitters were respanned and the steam flow circuitry rescaled. Independent testing is being performed to enable these measures to be implemented for Unit 1. The event is being reported, pursuant to 10 CFR 50.73(a)(2)(i)(B).

1.0 DESCRIPTION OF THE EVENT

On July 24, 1991, with Unit 1 and Unit 2 operating at 100% power, it was determined after analyzing recent special test data that the scaling for the Unit 2 Main Steam (MS) flow transmitters {EIIS-SB,FIT} was incorrect. These flow transmitters provide input to certain Reactor Protection System (RPS) {EIIS-JC} and Engineered Safety Features (ESF) actuation {EIIS-JE} signals. It was determined that the installed scaling factors resulted in setpoints in excess of the maximum allowed value of 110% (at full load) of full steam flow specified in Technical Specification (TS) 3.7.D (Table 3.7-4, Item 5).

The special test began on July 4, 1991 and was governed by procedure 2-ST-302. The objective of the test was to collect data at various power levels to verify the manufacturer's calibration curves for feedwater flow venturis {EIIS-SJ} 2-FW-FE-2476, 2-FW-FE-2486, and 2-FW-FE-2496, which were installed during the Unit 2 Cycle 10 refueling outage. Data collected in this test was used to perform final scaling calculations for the feedwater and steam flow transmitters. The data collected during the special test was discussed by telephone with NRC staff members on July 12, 1991.

Following data reduction and analysis, it was determined that the Unit 2 high steam flow setpoint for ESF actuation was non-conservative due to incorrect scaling. It was also determined that the steam flow setpoint for the steam flow/feedwater flow mismatch reactor trip was non-conservative due to incorrect density compensation and incorrect scaling. The nonconservatism in the steam flow/feedwater flow mismatch reactor trip was insufficient to exceed the value for reactor trip protective instrumentation settings in Technical Specification 2.3.A.3.c.

The event is being reported, pursuant to 10 CFR 50.73(a)(2)(i)(B), since this condition is prohibited by TS 3.7.D (Table 3.7-4, Item 5).

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

MS flow signals are used as an input to two of the plant's protective functions. High steam line flow coincident with either low Reactor Coolant System (RCS) {EIIS-AB} average temperature (Tave) or low steam line pressure results in a signal to the ESF protection logic, initiating Safety Injection (SI) and MS line isolation signals. Steam flow is also an input to the RPS logic, initiating a reactor trip, if the system senses a steam flow/feedwater flow mismatch condition coincident with low steam generator {EIIS-SG} water level in any steam generator.

The limiting steam line break analyses which take credit for the high steam flow channels are not sensitive to the steam flow setpoint. In the accident scenario, steam flow rapidly increases to a value well in excess of the setpoint. The Surry evaluation for small steam line breaks shows that break sizes in excess of 0.19 sq. ft. per loop would exceed the steam flow setpoint. The limiting Surry break size is 1.4 sq. ft., well in excess of this value. Therefore, the limiting steam flow used in the accident analysis will exceed the setpoint by several hundred percent. The recent review of scaling calculations indicated that the maximum error results in a worst case steam flow of 115% vice 110% allowed by TS for generating the ESF signal. The Surry evaluation also shows that, for smaller break sizes where the steam flow setpoint cannot be relied upon, adequate protection is provided by diverse channels (i.e., high containment pressure, low RCS pressurizer {EIIS-PZR} pressure).

The steam flow/feedwater flow mismatch reactor trip is a diverse, anticipatory trip which backs up the low-low steam generator level reactor protection function with one exception. The exception is a partial loss of feedwater due to failure of the steam generator level channel feeding the level control system {EIIS-LIC}. In this scenario, the two unaffected steam generators are adequate to remove the core heat until the reactor trip occurs on low-low steam generator level in the affected steam generator, high RCS pressurizer level, or excessive difference between RCS cold leg and hot leg temperatures (delta-T). In conjunction with the reactor trip, auxiliary feedwater is initiated on a low-low steam generator level signal prior to the onset of significant RCS heat-up. In addition, the nonconservatism in the steam flow/feedwater flow mismatch reactor trip was not sufficient to exceed the value for reactor trip protective instrumentation settings in Technical Specification 2.3.A.3.c.

Thus, the observed steam flow measurement uncertainties did not invalidate the accident analyses and the event has presented no adverse consequences to public health and safety.

3.0 CAUSE OF THE EVENT

In 1977 the main steam flow transmitter spans were changed on both Unit 1 and Unit 2 using an incorrect scaling methodology. The respanning was undertaken to correct an observed difference in the values for steam flow and feedwater flow wherein the steam flow was reading higher than the feedwater flow. Based on evaluation of the special test 2-ST-302 data, it is now known that the difference in the steam flow and feed flow measurements observed in 1977 was due to an error in the steam flow signal multiplier/divider scaling which caused an error in the steam flow indication.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

On July 22, 1991 for Unit 2 and July 24, 1991 for Unit 1, voltage biases were placed on the uncompensated steam flow signals. Based on the data obtained from 2-ST-302, this action reduced the Hi Steam Flow ESF actuation on Unit 2 below 110%. Since the same incorrect scaling information was utilized on Unit 1 in 1977, the voltage biases were also placed on the Unit 1 signals to effectively reduce the 1977 scaling error, thereby reducing the Hi Steam Flow ESF actuation value.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

Following an engineering evaluation of the special test data obtained from 2-ST-302, the Unit 2 steam flow and feedwater flow transmitters were respanned and the steam flow circuitry and associated Hi Steam Flow ESF actuation functions were rescaled in September 1991. These actions corrected the steam flow/feedwater flow mismatch to acceptable values.

The Unit 2 scaling effort was reviewed for applicability to Unit 1. From this review, it was determined that Unit 1 testing would be required. Therefore, independent testing is being performed for Unit 1 in conjunction with the 1992 refueling outage. The test results will be evaluated and, as appropriate, the Unit 1 steam flow and feedwater flow transmitters will be respanned and the steam flow circuitry and associated Hi Steam Flow ESF actuation functions rescaled.

6.0 ACTIONS TO PREVENT RECURRENCE

A scaling program is being undertaken to verify the scaling for ESF and RPS instrument loops. The program includes the development of a standard scaling methodology and scaling procedure. The new scaling process will then be utilized to perform the calculations for an instrument loop. Any enhancements identified in performing the first calculation will be incorporated into the methodology. The verification of the scaling for the ESF and RPS instrument loops will then be coordinated with the development of upgraded calibration procedures being performed under the Technical Procedures Upgrade Program.

7.0 SIMILAR EVENTS

None

8.0 MANUFACTURER/MODEL NUMBER

N/A