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

June 14, 1989

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

Serial No.: 89-020
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 Licensee Event Report for Units 1 & 2.

REPORT NUMBER

89-020-00

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,

fr

M. R. Kansler
Station Manager

Enclosure

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

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1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Surry Power Station, Units 1 and 2

DOCKET NUMBER (2): 0 5 0 0 0 2 8 0

PAGE (3): 1 OF 0 6

TITLE (4): Potentially Inoperable Reactor Protection Channel Due to High Leakage Currents in Cable While in Harsh Containment Environment

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 5	1 5	8 9	8 9	0 2 0	0 0	0 6	1 4	8 9		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 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)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)	X	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: M. R. Kansler, Station Manager

TELEPHONE NUMBER: AREA CODE 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	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)

During a review of instrument loop accuracies conducted in response to an NRC Environmental Qualification finding at North Anna Power Station, it was determined that the leakage current in cables manufactured by Continental Cable is excessive. This high leakage current could have potentially masked the Reactor Trip signal for Steam Generator (S/G) Lo Lo Level and the Safety Injection signal for Pressurizer Low Pressure, and prevented the required protective action from occurring during a harsh containment environment condition. Therefore, a potential unanalyzed condition could have existed. A four-hour non-emergency report was made to the NRC on May 15, 1989. Previous calculations of instrument accuracy did not consider the total loop errors introduced by high leakage current resulting from the low resistance values of this type of electrical cabling. The affected cabling is being replaced in both units.

1.0 Description of the Event

During a review of instrument loop accuracies being conducted in response to an NRC Environmental Qualification finding at North Anna Power Station, it was determined that the leakage current in cables manufactured by Continental Cable (EIIS-CBL1) could exceed required Reactor Trip and Safety Injection (SI) (EIIS-JE) actuating level currents. This cabling was used in the reactor protection instrument channels for the Steam Generator (S/G) (EIIS-HX) Narrow Range Level and Pressurizer (EIIS-PZR) Pressure protection circuits. This high leakage current could have potentially masked the Reactor Trip Signal for S/G Low Low Level and the Safety Injection signal for Pressurizer Low Pressure, and could have prevented the required protective action from occurring. The leakage is generated as a result of the low insulation resistance (IR) values during Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB) conditions in containment.

These high current leakage induced errors occur only under the harsh environments created during a LOCA or MSLB condition and do not affect instrument performance under non-accident conditions. A potential unanalyzed condition could have existed wherein the required Reactor Trip, or Safety Injection signals may not have occurred when required. A four-hour non-emergency report was made to the NRC on May 15, 1989, pursuant to 10CFR50.72(b)(2)(i).

2.0 Safety Consequences and Implications

The purpose of the low pressurizer pressure SI is to mitigate the consequences of a LOCA (primary break) and a MSLB (secondary break) by injecting borated water into the RCS. The low S/G level reactor trip is designed to mitigate the consequences of a loss of heat sink, which includes a feed line break in containment (secondary break).

Several indications are available in the control room to alert operators of the onset of these events. The parameters affected by a primary break in containment are containment temperature, containment pressure, containment sump level, containment radiation, reactor coolant makeup flow, RCS pressure, and pressurizer level. Parameters affected by a secondary break in containment are containment temperature, containment pressure, feedwater flow, steam flow, steam generator levels, RCS pressure, and pressurizer level and reactor coolant makeup flow. Changes to the above parameters would have been noted soon after the initiation of a small primary or secondary break of any significance. Control room annunciator alarms would have alerted the operator to any deviation from normal conditions of the above parameters except containment temperature. Annunciator procedures exist to provide operators guidance in actions required had abnormal indications been noted. In addition, operators have received extensive simulator training and are required to maintain the ability to quickly recognize and respond to these events. If protective actions are required, the operators have been instructed to manually initiate these actions and not to depend solely on the automatic initiation.

Large primary or secondary breaks in containment would rapidly increase containment pressure to the point where a high Consequence Limiting System (CLS) SI would be initiated. This would occur at a pressure of 17.7 psia in containment. In addition, a large steam line break in containment would result in the initiation of SI due to a steam header pressure greater than any steam line pressure by 120 psi.

Following the initiation of the above protective actions, operators would have implemented the Emergency Operating Procedures (EOP). These procedures are used to guide operators in taking the appropriate actions to place the plant in a safe and stable condition following an accident. Pressurizer pressure and S/G levels are some of the parameters monitored during the implementation of the EOPs. Specific procedures require action to be taken in response to these indications; consequently, erroneous indications could affect the optimum implementation of the EOPs. However, the total error generated in the instrumentation loops due to the harsh containment conditions could result in indicated values of greater than or less than the actual value.

Operators would have been alerted to the erroneous conditions by reviewing overall plant status and would monitor other indications to obtain information for that parameter. For pressurizer pressure and S/G narrow range levels, unaffected indications of RCS wide range pressure channels and S/G wide range level channels existed to provide operators with the information necessary to successfully implement the EOPs. In the event erroneous S/G level indication results in underfeeding the generators, Function Restoration Procedures and other guidance exist to maintain the critical safety functions. Examples are safety injection initiation, reactor coolant pump trip on loss of subcooling, inadequate core cooling procedures on high core exit thermocouple or low reactor level indication. These contingency actions have been designed to ensure that the applicable critical safety functions are maintained. Therefore, the health and safety of the public were not affected.

3.0 Cause

Previous calculations of instrument accuracy did not consider the total loop errors introduced by high leakage current resulting from the low IR values of this type of electrical cabling. Continental cabling has been environmentally qualified as documented in plant Qualification Documentation Review (QDR) packages.

The previously determined IR values for this cable were satisfactory up to containment temperatures of 280 degrees Fahrenheit. However, recalculation of total loop IR values; including cabling, containment penetration resistances, and instrument seal resistances, resulted in the total loop error in excess of 25% when added to transmitter and rack errors.

4.0 Immediate Corrective Action(s)

Since both units were in cold shutdown condition at the time of discovery, no immediate corrective actions were necessary.

5.0 Additional Corrective Action(s)

A review of instrumentation loops which utilize Continental cabling in reactor trip and SI circuits, as well as certain post-accident instrumentation, has been conducted. The following transmitters, cables, and channel functions are adversely affected by this leakage current:

UNIT(S)	DESCRIPTION
1 and 2	A, B, C S/G Narrow Range Level Channels I, II, III.
1 and 2	Pressurizer Pressure Protection Channels I, II, III.
1 and 2	Pressurizer Level Protection Channels I, II, III.
1 and 2	Reactor Coolant System (RCS) Wide Range Pressure (PT-1402 and PT-2402).

The affected cabling for the above instrumentation is being replaced in both units. The new cabling being installed is either Brand Rex or Rockbestos Firewall III XLPE cable. This cabling has been previously accepted for EQ as documented in station QDR files. The cable replacement will be completed prior to restart of the respective unit.

Loop accuracy reviews have been performed for the instrumentation identified in the Emergency Operating Procedures (EOP). Although these reviews did not consider cable induced errors, this effect has been addressed for the EOP loops as identified above in an ongoing engineering review. For the EOP loops located outside containment, the potential for an IR induced problem is considered low because:

a. The temperature profiles are not as severe resulting in higher insulation resistance values.

b. The EOP instrument is not normally required to function during the event which creates the harsh environment.

This ongoing engineering review will be completed prior to the end of 1989.

6.0 Action(s) Taken to Prevent Recurrence

None required.

7.0 Similar Events

None

8.0 Manufacturer/Model Number(s)

Continental Cable
Type XLPE/CSPE
NUS-341, 341A, and 411
P.O. 285, 1285, 1439 and 1458