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VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION
P. O. BOX 402
MINERAL, VIRGINIA 23117

10 CFR 50.73

March 22, 1993

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

NAPS:MPW
Docket Nos. 50-338
50-339
License Nos. NPF-4
NPF-7

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Units 1 & 2.

Report No. 50-338/93-006-00

This Report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Corporate Management Safety Review Committee for its review.

Very Truly Yours,

G. E. Kane
Station Manager

Enclosure:

cc: U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W.
Suite 2900
Atlanta, Georgia 30323

Mr. M. S. Lesser
NRC Senior Resident Inspector
North Anna Power Station



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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 WITH 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) |
|---|---|---|
| | 0 5 0 0 0 3 3 8 | 1 OF 0 4 |

TITLE (4) PREVIOUSLY UNIDENTIFIED POTENTIAL POST LOSS OF COOLANT ACCIDENT, EMERGENCY CORE COOLING SYSTEM LEAKAGE PATH TO ENVIRONMENT DURING RECIRCULATION MODE OF OPERATION DUE TO INADEQUATE INITIAL 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 NAMES | DOCKET NUMBER(S) |
| 0 2 | 2 5 | 9 3 | 9 3 | - 0 0 6 | - 0 0 | 0 3 | 2 2 | 9 3 | | 0 5 0 0 0 3 3 9 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): 6

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) | ☐ 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 |
|---|---|
| G. E. Kane | AREA CODE 7 0 3 — 8 9 4 - 2 1 0 1 |

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) ☒ NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

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


On February 25, 1993, at 1954 hours with Unit 1 defueled and Unit 2 in Mode 1 (100 percent power) a four hour report was made to the NRC pursuant to 10CFR50.72(b)(2)(iii)(C) due to the confirmation of a previously unidentified potential post-LOCA, Emergency Core Cooling System (ECCS) leakage path to the environment during the recirculation mode of operation. Failure of the check valve between the volume control tank (VCT) and the suction of the High Head Safety Injection (HHSI) charging pumps could result in the pressurization of the seal water heat exchanger by the low head safety injection (LHSI) pump discharge. The relief valve on this line discharges to the VCT which in turn relieves to the liquid waste system which could ultimately overflow to the auxiliary building sump. This condition is reportable pursuant to 10CFR50.73 (a)(2)(v)(C).

The cause of the condition was an inadequate initial design which failed to identify the potential leak path.

No significant safety consequences resulted from the condition. Compensatory actions to prevent a release to the environment are available. The ECCS operability is not compromised. Therefore, the health and safety of the public were not affected at any time during this condition.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

**LICENSEE EVENT REPORT (LER)**
**TEXT CONTINUATION**

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH 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.

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

## 1.0 Description of the Event

On February 25, 1993, at 1954 hours with Unit 1 defueled and Unit 2 in Mode 1 (100 percent power) a four hour report was made to the NRC pursuant to 10CFR50.72(b)(2)(iii)(C) due to the confirmation of a previously unidentified potential post-LOCA, Emergency Core Cooling System (ECCS) (EIIS System Identifier JE) leakage path to the environment during the recirculation mode of operation. Westinghouse Electric Corporation provided a Nuclear Safety Advisory Letter to Virginia Electric and Power Company concerning the potential effects of check valve (EIIS System Identifier CB, Component Identifier BFP) leakage during a LOCA. The advisory letter identified the potential for the check valve, located on the combined Volume Control Tank (VCT) (EIIS System Identifier CB, Component Identifier TK) and Seal Water return header (EIIS System Identifier CB) to the suction of the HHSI charging pump suction header (EIIS System Identifier BQ, Component Identifier P), to experience back leakage following a intermediate break LOCA. The check valve leakage would allow pressurization of the seal water heat exchanger (EIIS System Identifier CB, Component Identifier HX) which may result in subsequent relief valve (EIIS System Identifier CB, Component Identifier RV) operation allowing potentially contaminated containment sump water to enter the VCT during the recirculation phase of an intermediate break LOCA.

Subsequently, the VCT would fill and the VCT relief valve would operate resulting in a release of contaminated water to the waste drain system (EIIS System Identifier WH) and without any operator action would eventually provide a fission product release to the environment through monitored path ways.

The necessary isolation function performed by this check valve requires that it meet the active failure criteria. Leakage through the check valve/relief valve pathway involves only the single failure of an active component and, without operator intervention, could lead to a release to the environment. The failure will not prevent the SI system (EIIS System Identifier BQ) from completing its design function. The potential leakage would only occur when the SI pumps were aligned in the recirculation mode with a header pressure greater than or equal to the relief valve setpoint.

## 2.0 Significant Safety Consequences and Implications

The potential failure of the check valve poses no challenge to the reactor damage accident mitigation capability of the safety injection system (EIIS System Identifier BP) because injection flow requirements at the time of swap over are significantly less than the initial flow requirements. Also, the potential flow through the relief valve is a small fraction of LHSI flow capabilities.

Plant monitoring capability is unaffected. Available options for isolating leakage, in the unlikely event of a simultaneous intermediate break LOCA and check valve failure, ensure that no release to the environment would occur

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

2.0 Significant Safety Consequences and Implications (continued)

even if the check valve fails. Adequate time would be available to mitigate any release to the environment.

The leak rate for the Unit 1 check valve has been quantified. Based on that leak rate, it would take approximately 1.5 to 4 days to cause an actuation of the VCT relief valve. As such, by the time the level of the VCT becomes a problem the plant would be well into the recovery mode actions and alarms would alert operations personnel in the control room to both the rising VCT pressure and level.

In addition, if check valve leakage is low, as measured on Unit 1 and expected on Unit 2, then no challenge will result since the Station Emergency Operating Procedures will result in reducing the reactor coolant system (RCS) pressure. This RCS pressure reduction will result in the pressure at the seal water heat exchanger relief valve being reduced to less than its setpoint prior to filling the VCT.

A safety Evaluation was also performed and it was determined that this condition did not result in or constitute an unreviewed safety question.

3.0 Cause of the Event

The cause of the condition was an inadequate initial design which failed to identify the potential leak path.

4.0 Immediate Corrective Actions

Upon confirmation of a potential post-LOCA, Emergency Core Cooling System (ECCS) leakage path to the environment during the recirculation mode of operation, a four hour report was made to the NRC pursuant to 10CFR50.72(b)(2)(iii)(C).

5.0 Additional Corrective Actions

A Justification for Continued Operation (JCO) for both Units 1 and 2 was completed and provided the required compensatory actions.

6.0 Actions to Prevent Recurrence

An engineering evaluation will be performed to determine the actions necessary for the long term solution.

The Virginia Electric and Power Company's Industry Operating Experience Review program will determine if there is a need for additional action.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION



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

7.0 Similar Events

Licensee Event Report N1/2-89-013-00 identifies an unanalyzed condition regarding small break LOCA resulting from the Incore Flux Mapping Frame Assembly being unrestrained. During a seismic event the unrestrained frame could damage the incore flux mapping tubes located directly above the seal table.

8.0 Additional Information

None