Vepco

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

10 CFR 50.73

June 6, 1991

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

Serial No. N-91-011
NAPS:WCH
Docket Nos. 50-338
License Nos. NPF-4

Dear Sirs:

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

Report No. 91-011-00

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

9106130316 910606
PDR ADOCK 05000338
S PDR

JE22

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): NORTH ANNA POWER STATION UNIT 1

DOCKET NUMBER (2): 0 5 0 0 0 3 3 8

PAGE (3): 1 OF 0 4

TITLE (4): UNIT SHUTDOWN DUE TO EXCEEDING TECHNICAL SPECIFICATION LIMIT FOR REACTOR COOLANT SYSTEM PRESSURE BOUNDARY LEAKAGE

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 5	1 1	9 1	9 1	— 0 1 1 —	0 0	0 6	0 6	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 3 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
G. E. Kane, Station Manager	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	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)

At 0400 hours on May 11, 1991, with Unit 1 at 30 percent power, a containment entry team identified a source of leakage between the "B" cold leg loop stop valve and the isolation valve for the 3/4 inch (3/8 inch inside diameter) upper disc pressurization line, and the Action Statement of Technical Specification 3.4.6.2 was entered. A unit shutdown commenced at 0500 hours due to an unisolable Reactor Coolant System (RCS) pressure boundary le[illegible]. A Notification of Unusual Event (NOUE) was declared in accordance with the emergency plan, and all appropriate notifications were made in a timely manner. The unit was placed in cold shutdown (mode 5) at 1640 hours. This event is reportable pursuant to 10CFR50.73 (a)(2)(i)(A) as a completion of a plant shutdown required by TS. A one hour report was made pursuant to 10CFR50.72 (a)(i).

A visual inspection of the pipe crack indicated a probable high cycle fatigue failure mechanism. No significant safety consequences resulted from the pipe crack because the leak, approximately 0.7 gpm, was much less than the normal charging system make up capacity. In addition, there are no pipe whip or impingement concerns associated with the disc pressurization piping. The health and safety of the public was not affected at any time during this event because the leak was confined to the containment structure and no release was made to the environment.

1.0 Description of the Event

At 1925 hours on May 9, 1991, a unit 1 ramp down from 98 percent power to 30 percent power was initiated to reduce exposure to personnel attempting to locate and repair a 0.40 gpm source of unidentified leakage. Leakage was noted in the "B" reactor coolant pump (RCP) (EIIS System identifier AB, Component identifier P) cubicle which was later identified as two Resistance Temperature Detector manifold valves. Subsequently, the valves were repaired and a RCS leak rate test was performed. The results of the test indicated an increase in the RCS leak rate to 0.7 gpm, and additional walkdowns were performed to identify the source of the leakage. A containment entry made on May 10 at at 2350 hours revealed leakage in the vicinity of the "B" cold leg loop stop valve (EIIS System Identifier AB, Component Identifier ISV); however, its exact point source could not be detected. On May 11 at 0400 hours it was determined after removal of insulation that the leakage source was a RCS pressure boundary which could not be isolated or repaired at the current operating conditions, and the Action Statement of Technical Specification (TS) 3.4.6.2 was entered.

On May 11, 1991, at 0500 hours a Unit 1 shutdown from 30 percent power commenced due to Reactor Coolant System (RCS) pressure boundary leakage from the 3/4 inch (3/8 inch inside diameter) upper disc pressurization line for the "B" cold leg loop stop valve. The amount of the leakage was measured at approximately 0.7 gpm. The Emergency Plan Implementing Procedures (EPIP) were entered and a Notification of Unusual Event (NOUE) was declared in accordance with EPIP-1.01, Tab B-4, which specifies declaration of a NOUE for events involving leakage from the RCS which require unit shutdown. The unit was placed in cold shutdown (mode 5) at 1640 hours, and the NOUE was terminated. This event is reportable pursuant to 10CFR50.73 (a)(2)(i)(A) as a completion of a plant shutdown required by Technical Specifications. A one hour report was made pursuant to 10CFR50.72 (a)(i)

2.0 Significant Safety Consequences and Implications

No significant safety consequences resulted from the pipe crack because the leakage was identified promptly and appropriate actions were taken to bring the unit to shutdown conditions. If the crack had developed and propagated quickly prior to cooldown and depressurization and a complete line failure occured, the condition would have been bounded by the small break loss-of-coolant accident (SBLOCA) analysis. The health and safety of the public was not affected at any time during this event because the leak was confined to the containment structure and no release was made to the environment.

3.0 Cause of the Event

A preliminary analysis of the pipe using transmission electron microscopy (TEM) determined the cause of the failure to be high cycle fatigue. The 180 degree indication was observed to have approximately 5 percent of fresh growth, and several indications of load change were identified based on "beach marks" on the fracture surface..

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
NORTH ANNA POWER STATION UNIT 1	0 5 0 0 0 3 3 8	9 1	0 1 1	0 0	0 3 OF 0 4

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

3.0 Cause of the Event (continued)

Engineering Mechanics reviewed the piping configuration and stress analyses for the portions of three lines in question for Units 1 and 2 and determined that stress levels in the piping system are relatively low and well within the code allowances. This level of cyclic stress, due to a variation of loads during normal/upset plant events, would indicate that fatigue of an integral connection would not be expected. A fatigue related failure in the pipe can only be attributed to some externally induced vibration to which the piping responded at a higher mode leading to a high cycle fatigue failure. It appears that a high frequency and low amplitude vibration initiated a crack and caused it to propagate slowly to develop into a through wall crack over a long period of time as evidenced from the development of a detectable leak. This type of unanticipated loading condition is outside the normal design practice. Therefore, a catastrophic failure of this piping is not expected to occur without first leaking allowing prompt corrective measures.

4.0 Immediate Corrective Actions

Several teams had made entries into the containment building to identify and repair leakage. The Action Statement of TS 3.4.6.2 was entered at 0400 hours upon determining that the leakage was RCS pressure boundary leakage. At 0500 hours a Unit 1 shutdown from 30 percent power commenced. The Emergency Plan Implementing Procedures (EPIP) were entered and a Notification of Unusual Event (NOUE) was declared due to RCS pressure boundary leakage which required unit shutdown. Initial notification to the State and Local jurisdictions was completed at 0506 hours. The initial notification to the NRC was completed at 0508 hours. The unit was placed in cold shutdown (mode 5) at 1640 hours, and the NOUE was terminated.

5.0 Additional Corrective Actions

The RCS was drained down to reduced inventory conditions to support maintenance on the failed pipe. The complete section of the upper disc pressurization line piping for the "B" cold leg loop stop valve was replaced. The other eleven disc pressurization lines for Unit 1 were inspected using liquid penetrant techniques. Two of the lines were determined to require replacement due to linear indications, and the piping was sent to a Westinghouse laboratory for analysis. Indications on three other welds were removed through light grinding and welding. Six remaining lines were found in satisfactory condition.

6.0 Actions to Prevent Recurrence

Technical Specification surveillance 4.4.6.2.1.d states that the RCS leakage rates shall be demonstrated less than the limits by performing a RCS water inventory every 72 hours during steady state operation. This surveillance frequency has been increased administratively to every 24 hours on Units 1 and 2 and will continue until additional analysis and inspections are completed. If a confirmed leak rate increase of 0.15 gpm over the baseline occurs, inspections will be performed to identify the leakage source.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
NORTH ANNA POWER STATION UNIT 1	0 5 0 0 0 3 3 8	9 1	— 0 1 1 —	0 0	0 4 OF 0 4

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

6.0 Actions to Prevent Recurrence (continued)

The disc pressurization piping for Unit 1 will be inspected during the next refueling outage, and the same piping for Unit 2 will be inspected during the next cold shutdown.

7.0 Similar Events

LER N1-86-013-00 documents a Unit 1 rampdown to hot standby (mode 3) due to exceeding the allowable TS limit for unidentified leakage on August 13, 1986. The source of the leak was later identified as the "C" loop Resistance Temperature Detector bypass flow element.

LER N1-87-017-00 documents a Unit 1 manual trip and subsequent automatic safety injection system initiation due to a steam generator tube rupture in the "C" steam generator on July 15, 1987.

8.0 Additional Information

All repairs to the Unit 1 loop stop valve disc pressurization line welds were completed at 1320 hours on May 19, 1991. The RCS was filled and vented by 1023 hours on May 19, and the unit entered mode 3 at 0946 hours on May 20. An RCS leak rate determination was performed at 0154 hours on May 21 with an RCS unidentified leakage of essentially 0.0 gpm.

North Anna Unit 2 was in mode 1 at 100% power throughout this event and was not affected. However, increased frequency of RCS leak rate tests will continue until additional analysis and inspection of the subject piping is completed.