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

10 CFR 50.73

February 19, 1992

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

Serial No. N-92-03
NAPS:WCH
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 and 2.

Report No. 50-338/92-003-00

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

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 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 Units 1 and 2

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

PAGE (3): 1 OF 0 5

TITLE (4): RESIDUAL HEAT REMOVAL SYSTEM OVERPRESSURE PROTECTION

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 1	2 1	9 2	9 2	- 0 0 3	- 0 0	0 2	1 9	9 2	North Anna Unit 2	0 5 0 0 0 3 3 9
										0 5 0 0 0

OPERATING MODE (9): 5

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)	X 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: G. E. Kane, Station Manager

TELEPHONE NUMBER: 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)

On January 21, 1992, with Unit 1 in Mode 5 and Unit 2 in Mode 1, an Engineering evaluation performed in response to Westinghouse letters VRA 90-544 and VRA 90-545 determined that the Residual Heat Removal (RHR) System suction relief valve discharge piping arrangement may not pass its design flow rate to protect the RHR system from overpressurization when it is not isolated from the Reactor Coolant System at or near 350°F during a charging/letdown mismatch event. This event is reportable pursuant to 10CFR50.73 (a)(2)(v)(B) as a condition that alone could have prevented the fulfillment of the safety function of a system that is needed to remove residual heat. A four hour report was made pursuant to 10CFR50.72 (b)(2)(iii)(B).

The cause of the event was a potential design deficiency of the RHR System suction relief valve discharge piping arrangement.

No significant safety consequences would result from this event beyond those analyzed in the UFSAR. The UFSAR evaluated a break caused by an overpressurization event in the largest RHR line that could adversely impact both RHR trains simultaneously. Results of the analysis confirm that the makeup required to preclude an unsafe condition can be provided. Therefore, the health and safety of the public were not affected at any time due to this event.

1.0 Description of the Event

On January 21, 1992, with Unit 1 in Mode 5 and Unit 2 in Mode 1, an Engineering evaluation performed in response to Westinghouse letters VRA 90-544 and VRA 90-545 determined that the Residual Heat Removal (RHR) System (EIIS System Identifier BP) suction relief valve discharge piping arrangement (Component Identifier RV) may not pass its design flow rate during a charging/letdown (EIIS System Identifier CB) mismatch event to protect the RHR system from overpressurization when it is not isolated from the Reactor Coolant System (RCS) (EIIS System Identifier AB) at or near 350° F. This event is reportable pursuant to 10CFR50.73 (a)(2)(v)(B). A four hour report was made pursuant to 10CFR50.72 (b)(2)(iii)(B).

Virginia Power was notified by Westinghouse (W120) on February 21, 1990, (Draft Letter 01/23/90 with Follow-up 02/21/90) of the discovery of generic inconsistencies and misunderstanding regarding the design basis and the mechanical capabilities of the RHR System suction piping relief valves. Westinghouse stated that the RHR relief valves may not be able to achieve their design rated capacities if the actual backpressure in the discharge piping exceeds the relief valve's allowable backpressure limit. An initial engineering evaluation of the problem completed June 11, 1990, determined that the problems described in the Westinghouse letter did not apply to North Anna Units 1 and 2. A subsequent independent Engineering review of the problem completed January 21, 1992, determined that the original design basis relief flow capacities may not be met. The problem concerns the relief capacity of the relief valves in mitigating an RHR overpressurization event when the temperature of the RCS is high enough to cause flashing of the water being discharged by the RHR suction relief valves (two phase discharge).

The North Anna Units 1 and 2 RHR Systems each have a 600 psig pressure rating of the piping, and components are designed to operate at less than 350° F and 450 psig. Station Operating procedures and system interlocks limit the operation of the system to 350° F and 418 psig. Each RHR System has two relief valves whose original design basis was to provide RCS overpressure protection when the RHR System is in operation during a charging/letdown mismatch event. The two RHR relief valves would begin to discharge at a 467 psig set pressure and would pass a maximum of 900 gpm at 514 psig to the pressurizer relief tank (EIIS System Identifier CA, Component Identifier TK) which is normally maintained at 3 to 5 psig. Each relief valve has a maximum design backpressure of 5 psig and the capacity to deliver its design basis flow rate; however, it may be prevented from doing so due to flow resistance in the piping between the relief valve and the PRT.

Since North Anna Units 1 and 2 were originally built, the Low Temperature Overpressure Protection (LTOP) system was installed in each unit. This system's design utilizes the Pressurizer Power Operated Relief Valves (EIIS System Identifier AB, Component Identifier PZR-RV) to mitigate the consequences of a worst case RCS overpressure transient assuming a net mass

1.0 Description of the Event (continued)

addition equivalent to one charging pump's flow. The Technical Specification minimum temperature at which two charging pumps can be operated is > 324° F for Unit 1 and > 340° F for Unit 2. Currently, the LTOP system is activated when the RCS temperature is ≤ 261° F (Unit 1) and ≤ 340° F (Unit 2). The LTOP system was not originally designed to protect the RHR system; however, it can provide RHR system overpressure protection when LTOP is placed into service and a maximum of one charging pump is available.

The Engineering study presents a concern regarding a loss of Instrument Air (IA) (EIIS System Identifier LD) accident. On a loss of IA, the charging flow control valve fails open while the letdown system valves fail closed. Assuming no operator actions, this charging/letdown mismatch accident could cause overpressurization of the RHR system between the LTOP setpoint temperature and 350°F with only one charging pump running.

2.0 Significant Safety Consequences and Implications

No significant safety consequences would result from this event beyond those analyzed in the UFSAR. The UFSAR evaluated (UFSAR 5.5.4.3.2) a break caused by an overpressurization event in the largest RHR line that could adversely impact both RHR trains simultaneously. Results of the analysis confirm that the makeup required to preclude an unsafe condition can be provided.

The occurrence of an RHR line break when in the RHR mode has been analyzed (UFSAR 5.5.4.3.1) for a postulated RHR moderate energy line break during shutdown. The analysis is conservatively based on the break occurring within four hours after reactor shutdown with the reactor coolant system at 450 psig and 350°F, and the pressurizer level at 21.4%. An assessment was also made to determine equipment necessary to mitigate a RHR line break to ensure that the core is again covered.

The analysis showed that the operator has 44 minutes after the initial alarm to take any appropriate action to ensure core immersion. The analysis further established that one charging pump will provide adequate flow to sustain the system in a safe condition, and an initial alarm signal (low-pressurizer-level deviation alarm) at 16.4% (5% below zero power programmed level of 21.4%) will occur within 30 seconds of the event initiation followed by another alarm (low level heater cutoff) at 15% and then another alarm (low level safety injection trip setpoint) at 5%. The analysis conservatively assessed the largest RHR line that could adversely impact both RHR trains simultaneously.

2.0 Significant Safety Consequences and Implications (continued)

Results of the analysis confirm that the required make-up can be provided by the inservice charging pump. Even if a 10 minute delay time for operator action and a single failure are assumed, an unsafe condition would not result. Specifically, 34 minutes should still remain available for the initiation and effective operation of necessary equipment. Moreover, it is only if the single failure assumption is invoked that operator action to start the backup charging pump would be necessary. The operator can initiate the starting of the pump from within the main control room and flow can be established within one minute. Primary coolant loss through the break will lower the level in the reactor vessel to the hot leg nozzle elevation, assuming no charging pump flow at 33 minutes from break initiation. The start of charging pump flow in 11 minutes will delay that time, and the level will stabilize at the hot leg nozzle level until the break is isolated. Following isolation of the break, the original pressurizer level will be reestablished within 75 minutes. Specifically, 34 minutes remain for operators, from within the control room, to start the backup charging pump (if required) and initiate closure of the RHR isolation valves. Following isolation of the break, the original pressurizer level will be reestablished within 75 minutes. This study is still bounding for this event and compensatory actions considered as a result of this study are still acceptable. Therefore, the health and safety of the public were not affected at any time due to this event.

3.0 Cause of the Event

The cause of the event was a potential design deficiency due to generic inconsistencies and misconceptions regarding the design basis and the mechanical capabilities of the RHR System suction relief valves.

4.0 Immediate Corrective Actions

A Standing Order has been implemented which directs Operations Department personnel to ensure the LTOP system is inservice at any time the RHR system is unisolated from the RCS. The RHR system will be maintained isolated until the RCS temperature is decreased to 323°F (Unit 1) or 339°F (Unit 2) and all but one charging pump is in pull-to-lock. In addition, direction is provided to ensure the AFW System is capable of providing sufficient inventory to cooldown the RCS until LTOP and RHR can be placed in service.

5.0 Additional Corrective Actions

A Technical Specification change package is being developed for both units which raises the temperature at which a maximum of one charging pump can be operated to > 350°F and requires that LTOP be in service whenever RHR is aligned to the RCS. Approval of this change eliminates the overpressurization concern of the RHR system.

Evaluations to determine if the RHR System has sufficient relief capacity will continue.

6.0 Actions to Prevent Recurrence

The standing order will remain in effect until the TS change is approved which resolves the overpressure concern.

7.0 Similar Events

None.

8.0 Additional Information

None.