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

December 23, 1991

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

Serial No. N-91-033
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 Voluntary Report Revision applicable to North Anna Units 1 and 2.

Report No. 50-338/91-020-01

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,

Original signed by
J. A. Stall
Assistant Station Manager NS&L for

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

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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)	PAGE (3)
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TITLE (4)
SERVICE WATER SYSTEM CONFIGURATION DURING LOGIC TESTING - VOLUNTARY REPORT

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)
1 0	3 1	9 1	9 1	- 0 2 0	- 0 1	1 2	2 3	9 1	NORTH ANNA UNIT 2	0 5 0 0 0 3 3 9
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 0 0	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)	X 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, 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)	NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	X				

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

At 1117 hours on October 31, 1991, with Units 1 and 2 operating at 100 percent power (Mode 1), it was discovered that the shared Service Water (SW) system may not have been able to provide design flow to an accident unit's Recirculation Spray Heat Exchangers. This discovery was made during the performance of a periodic test which removed the automatic start function of the Unit 2 Emergency Diesel Generator which provides emergency power to the Unit 2 "B" SW pump during an accident. The test also removed the automatic start function of the Unit 1 "B" SW pump. Operating procedures require that the SW pump discharge pressure be adjusted to 258 psig by manually throttling Component Cooling Heat Exchanger CCHX valves when less than 4 pumps are operable; however, the system was not throttled as required. It was initially determined that sufficient SW flow may not have been available to mitigate the consequences of an accident, and a four hour report was made on October 31, 1991, pursuant to 10 CFR 50.72 (b)(2)(iii)(D).

On November 26, 1991, an engineering calculation was completed and reviewed by the Station Nuclear Safety and Operating Committee (SNSOC) which determined that sufficient SW flow was available to mitigate the consequences of an accident. Therefore, the health and safety of the public were not affected during the event. The NRC was notified of the change in reportability on November 26, 1991, at 1240 hours. This voluntary report is being submitted to discuss the event and the basis for reclassification to nonreportable status.

NRC FORM 366 (5-92)

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 5/31/92

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.

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

1.0 Description of the Event

At 1117 hours on October 31, 1991 with Units 1 and 2 operating at 100 percent power (Mode 1), it was discovered that the shared Service Water system (SW) (EIIS System Identifier BI) may not have been able to provide design flow to an accident unit's Recirculation Spray Heat Exchangers (RSHX) (EIIS System Identifier BE, Component Identifier HX). This discovery was made during the performance of Periodic Test 2-PT-36.1B "Reactor Protection and ESF Logic Test Train B." The test blocked the automatic start function of the Unit 2 "J" Emergency Diesel Generator (EDG) (EIIS System Identifier EK, Component Identifier DG) which provides power to the Unit 2 "B" SW pump during an loss of power event. The test also defeated the automatic start function of the Unit 1 "H" SW pump. Current operating procedures require that the SW system flow resistance be adjusted to obtain a 58 psig pump discharge pressure by manually throttling Component Cooling Heat Exchanger (CCHX) (EIIS System Identifier CC) valves when less than 4 SW pumps are operable. This throttling ensures that the non-accident flow is limited to a value consistent with the previous flow balance initial conditions. Since the system was not throttled and only two pumps were operable, it was initially determined that sufficient SW flow would not have been available to mitigate the consequences of an accident. Therefore, a four hour report was made at 1510 hours on October 31, 1991 pursuant to 10CFR50.72(b)(2)(iii)(D)

TS 3.7.4.1 requires two operable SW loops to satisfy the single active failure requirements in 10 CFR 50 General Design Criteria 44. An unthrottled SW system requires three running pumps to meet this requirement, while only two running pumps are required when the system is throttled. When three SW pumps are operable, SW flow to the non-accident unit Component Cooling Heat Exchangers (CCHX) must be throttled to ensure that design flows are supplied to the accident unit RSHXs in the event that one of the three SW pumps fail. When only 2 SW pumps are operable and the SW system is not throttled, then a 72 hour LCO under TS 3.7.4.1 is entered because the SW system is vulnerable to a single failure. When operating under the LCO, no additional component failures need be considered.

If a Design Basis Accident (DBA) were to occur, a Containment Depressurization Actuation (CDA) would be initiated. If the SW system is aligned in an unthrottled condition with only two pumps running, the DBA unit RSHXs may not receive design flows. On November 26, 1991, an engineering calculation was completed and reviewed by the Station Nuclear Safety and Operating Committee (SNSOC) which determined that, in fact, sufficient SW flow would have been available to mitigate the consequences of an accident.

The NRC was notified of the change in reportability on November 26, 1991, at 1245 hours. This voluntary report is being submitted to discuss the event and the basis for reclassification to nonreportable status.

NRC Form 366A (5-92)

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

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.

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

2.0 Significant Safety Consequences and Implications

The operability of the shared SW system ensures that sufficient cooling capacity is available for safety related equipment during normal and accident conditions. During a design basis accident, both loops of SW cross connect to create a single large SW system. The affected units component cooling heat exchangers (CCHX) (EIIS System Identifier CC, Component Identifier HX) isolate to ensure that sufficient flow is provided to both the non-affected and affected unit's components.

With four service water pumps operable, the unthrottled flow resistance of the system is such that greater than design flows are achieved if a single pump or power supply failure occurs following an accident. When three service water pumps are operable, the flow resistance of the system is adjusted to ensure that design flows are achieved if a single pump or power supply failure occurs following an accident. When only two SW pumps are operable, the design basis condition can still be met provided that the flow resistance of the system is adjusted and no additional failures occur.

Operators in the control room recognized that the SW system was in a restricted condition after the automatic start function of the EDG was disabled and the automatic start function of the Unit 1 S.W. pump had been defeated. If a DBA had occurred during that time, operations personnel would have responded by placing the Unit 2 "J" EDG and the Unit 1 "B" SW pump in operation.

An engineering calculation (SE-0011) was reviewed by SNSOC on November 26, 1991 which determined that the shared SW system would have provided sufficient flow to either unit if an accident were to occur. This conclusion assumed no additional failures of a SW pump or EDG. Since sufficient SW flow would have been available to mitigate the consequences of an accident, the health and safety of the public was not affected during this event.

3.0 Cause of the Event

The condition was caused by personnel error in that the test procedure was inadequate. The procedure did not address the operating restrictions associated with the service water system.

A previous standing order outlining the service water system requirements was incorporated into the applicable station operating procedures; however, the engineering, electrical and instrumentation procedures, which may have a potential effect on SW system operability were not adequately screened for impact.

4.0 Immediate Corrective Actions

SW header "B" was throttled to greater than or equal to 58 psig and spray valves were opened.

NRC Form 366A (6-89)

NRC FORM 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

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-630), 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)			PAGE (3)		
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

4.0 Immediate Corrective Actions (continued)

The Unit 1 "B" SW pump was returned to automatic operation.

5.0 Additional Corrective Actions

Following completion of 2-PT-36.1B "Reactor Protection and ESF Logic Test Train B," the Unit 2 "3" EDG was returned to auto remote which enabled its automatic start circuit.

The "Reactor Protection and ESF Logic Tests" were revised to address the tests potential impact on the new SW system requirements.

A review of similar test procedures performed during 1991 was performed. During the review, similar operating conditions were identified as existing during tests on January 29, 1991, February 20, 1991, and October 4, 1991. These conditions were also evaluated and it was determined that sufficient flow would have been available for an accident in each case.

6.0 Actions to Prevent Recurrence

Following the October 31, 1991, event, a technical review of the population of procedures from the standpoint of surveillance and testing was performed to address the tests potential impact on the new SW system requirements. 22 procedures were identified as requiring revision, and the required SW system information has been incorporated into 16 of those. The remaining six revisions will be completed prior to their next performance and in all cases before April 30, 1992.

We believe that final resolution of the issue of [illegible]during adequate design bases flows to both units RSHXs during postulated events while still providing adequate operating margin for normal operations will be achieved during the upcoming refueling outages. Our present plans are to perform additional flow balance testing on both units. We would expect this testing to validate any combination of SW pump flow configurations, providing the UFSAR assumptions of only one CCHX in service per unit is adhered to.

7.0 Similar Events

Licensee Event Report (LER, 88-024-00, submitted for Units 1 and 2 November 15, 1988, documents SW flow not within Updated Final Safety Analysis Report (UFSAR) assumptions.

LER 89-008-00, submitted for Unit 1 May 12, 1989 and revised as LER 89-008-01 on June 23, 1989, documents SW flow to the RSHX's as less than design.

LER 90-012-00, submitted for Units 1 and 2 January 2, 1991, documents SW system operation in an unanalyzed condition causing possible low flow to RSHXs.

NRC Form 366A (6-89)

NRC FORM 366A (5/92) U.S. NUCLEAR REGULATORY COMMISSION

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

8.0 Additional Information

A TS change in the form of an amendment was submitted for Units 1 and 2 to address SW system operability requirements. The NRC issued amendment numbers 152 and 136 to facility operating license numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2 on December 13, 1991.

NRC Form 366A (5/92)