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FACIL:50-261 H.B. Robinson Plant, Unit 2, Carolina Power & Light C 05000261
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 89-011-00:on 891006,auxiliary feedwater sys flowrate could exceed limits of accident analysis.W/891106 ltr.
W/8 ltr.

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Serial: RNPD/89-3807
(10 CFR 50.73)

United States Nuclear Regulatory Commission
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
LICENSEE EVENT REPORT 89-011-00

Gentlemen:

The enclosed Licensee Event Report (LER) is submitted in accordance with 10 CFR 50.73 and NUREG-1022 including Supplements No. 1 and 2.

Very truly yours,

R E Morgan

R. E. Morgan
General Manager
Robinson Nuclear Project Department

CTB:lht

Enclosure

cc: Mr. S. D. Ebneter
Mr. L. W. Garner
INPO

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1/1

Enclosure to Serial: RN[illegible]/89-3807

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)	DOCKET NUMBER (2)	PAGE (3)
H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2	0 5 0 0 0 2 6 1	1 OF 0 4

TITLE (4)
AUXILIARY FEEDWATER SYSTEM FLOWRATE COULD EXCEED LIMITS OF ACCIDENT ANALYSIS

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	0 6	8 9	8 9	— 0 1 1 —	0 0	1 1	0 6	8 9		0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

OPERATING MODE (9): N

POWER LEVEL (10): 0 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)	[ ] 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	TELEPHONE NUMBER	
	AREA CODE	
C. T. Baucom, Senior Specialist	8 0 3	3 8 3 - 1 2 5 3

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)* [X] NO

EXPECTED SUBMISSION DATE (15) — MONTH / DAY / YEAR

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

With the plant in cold shutdown, calculations were performed in support of a Plant Modification to increase the diameter of the Auxiliary Feedwater (AFW) Pump suction piping. These calculations showed that a main steamline break concurrent with a loss of non-safety grade power to the Steam Driven (SD) AFW pump discharge flow control valve could result in exceeding the limits of the Main Steamline Break Accident Analysis. The results of these calculations were conveyed to site personnel at 1610 hours on October 6, 1989. At 1654 hours, pursuant to the requirements of 10CF50.72(b)(2)(i), a four hour non-emergency event notification was made to the NRC via the Emergency Notification System (ENS). To correct this situation, a Plant Modification was developed and implemented to decrease the flowrate setpoint for the discharge flow control valve and install a mechanical device to limit valve travel in the open direction. This modification will ensure that the valve operating range will comply with the limits provided in the accident analyses. This report is submitted pursuant to the requirements of 10CFR50.73(a)(2)(ii)(B).

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

I. Description

On August 22, 1989, Carolina Power and Light Company (CP&L) initiated a shutdown of H. B. Robinson Unit 2 (HBR2).[1] This shutdown was based on calculations which indicated that adequate Net Positive Suction Head (NPSH) could not be assured for all possible combinations of running Auxiliary Feedwater (AFW) Pumps and Condensate Storage Tank (CST) levels.[2] The cause of this situation was identified as a design deficiency in that the AFW pumps' suction piping was inadequate to provide the required NPSH under the postulated worst case conditions.[3]

During performance of calculations to support the increased suction pipe diameter, a condition was identified where the assumptions and criteria of the Main Steamline Break Accident Analysis could be exceeded. Specifically, the SD AFW pump discharge flow control valve, FCV-6416, is normally set to maintain a constant flow of approximately 600 gpm to the steam generators upon automatic start of the SD AFW pump. This valve receives power from a non-safety grade lighting panel, LP-26, and fails to the open position upon loss of power or loss of control signal. As described in the Updated Final Safety Analysis Report (UFSAR), the Main Steamline Break Accident Analysis assumes an upper limit of 1300 gpm AFW flow for a total time of ten (10) minutes. Assuming a loss of non-safety grade power during this accident scenario, the SD AFW pump would not have the capability for flow control. Therefore, the resulting SD AFW pump flow combined with the total flow of 650 gpm from both MD AFW pumps could not be assured to remain below 1300 gpm to the depressurized steam generator. This would create the potential to violate an accident analysis as provided in Chapter 15 of the UFSAR.

The results of these calculations were conveyed to site personnel at 1610 hours on October 6, 1989. At 1654 hours, pursuant to the requirements of 10CFR50.72(b)(2)(i), a four hour non-emergency event notification was made to the NRC via the Emergency Notification System (ENS).

II. Cause of Event

The cause of this condition is a design deficiency in that the failure mode of the discharge flow control valve created the potential for exceeding the criteria of an UFSAR Chapter 15 accident analysis.[4]

1 H. B. Robinson Steam Electric Plant, Unit No. 2 is a Westinghouse 700 MW Pressurized Water Reactor in commercial operation since March 1971.

2 Reference Licensee Event Report, LER 89-010-00, "Inadequate Auxiliary Feedwater Pump Net Positive Suction Head."

3 Cause Code: B

4 Cause Code: B

*U.S.GPO 1986-0-624-538/456

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)

III. Analysis of Event

As stated above, the calculations showed that the criteria of the Main Steamline Break Accident Analysis could be exceeded if a loss of non-safety grade power to the SD AFW pump discharge flow control valve was assumed in conjunction with a main steamline break. The specific AFW flow criteria which could be exceeded is the upper limit of 1300 gpm for a total time of ten minutes.

If this event and the concurrent failures were to occur, the Emergency Operating Procedures (EOP) provide instructions which should mitigate the consequences of such an occurrence. End Path Procedure, EPP-11, "Faulted Steam Generator Isolation," provides instructions for isolating all sources of feedwater and auxiliary feedwater to a faulted steam generator. Also, generic EOP instructions with regard to maintenance of non-faulted steam generator levels would ensure that flow to the steam generator(s) is maintained at some value less than the maximum available. These procedural actions should ultimately result in a moderation of overall flow, and should substantially reduce flow to the faulted steam generator. As such, it is reasonable to assume that the instructions provided within the EOPs would provide a mitigating effect on the consequences of such an event.

This event is reported in accordance with the requirements of 10CFR50.73(a)(2)(ii)(B).

IV. Corrective Actions

To ensure that the upper flow limit 1300 gpm will not be exceeded, a Plant Modification was developed and implemented to decrease the flowrate setpoint for the discharge flow control valve and install a mechnical device to limit valve travel in the open direction.[5] This modification will ensure that the maximum flowrate through the valve is 300 gpm with a loss of power to the valve actuator and all steam generators pressurized to 1133 psia. With this adjustment, the flowrate will be limited to 626 gpm in the event of a main steamline break and depressurization of a steam generator. This flow, when combined with the total flow from both MD AFW pumps, will remain below the 1300 gpm limit. The SD AFW pump will remain available to provide at least 240 gpm under all accident scenarios, which is within the lower limit specified in the accident analysis. This adjustment will also maintain the SD AFW pump flow within the limits of the manufacturer's pump curve.

5 Nuclear Plant Modification, MOD-1025, "Auxiliary Feedwater Steam Driven Pump Flow Control Valve Setpoint Change."

NRC Form 366A
(9-83)

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

In addition, a Plant Improvement Request has been initiated to determine and implement corrective actions to preclude the failure of the discharge flow control valve from adversely affecting AFW system operability and the Main Steamline Break Accident Analysis.[6] The permanent corrective actions are intended to result in the SD AFW pump discharge flow control valve setpoint returning to and remaining at 600 gpm.

Finally, the CP&L Nuclear Fuel Section (NFS) has reviewed the Main Steamline Break Accident Analysis to determine if refinements could be made to certain margins contained within the analysis. These reviews indicate that an upper flow limit of 1325 gpm is appropriate. This revised upper flow limit will provide additional margin between the plant setpoints and the criteria of the accident analysis.

V. Additional Information

A. Failed Component Identification

None

B. Previous Similar Events

None

6 PCN 89-211/00, "FCV-6416 Upgrade."