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INADEQUATE NPSH FOR AUXILIARY FEEDWATER PUMPS
EVENT FOLLOWUP REPORT 90-007
10 CFR 50.72 #16375
AUGUST 16, 1989
PLANT- H.B. ROBINSON
PROJECT MANAGER- R. LO
COGNIZANT ENGINEER- J. THOMPSON

PROBLEM
The licensee had determined that adequate net positive suction head (NPSH) pressure for the auxiliary feedwater (AFW) pumps could not be assured for all possible combinations of running AFW pumps and condensate storage tank (CST) levels.

CAUSE
A design deficiency in the AFW pumps' suction piping existed since initial plant startup. This event existed since construction in part due to a number of reasons, which are discussed, along with an explanation of root causes, below.

BACKGROUND
In October of 1987, the licensee initiated a safety system functional inspection (SSFI) on the AFW system. Information was collected during the month of December 1986. Shortly after the data collection, a reactor trip occurred in which a member of the SSFI team was present in the control room. The licensee initiated a special project report and concluded that an NPSH problem did not exist, but additional testing should be performed. Subsequently, two more reactor trips occurred, during which degraded AFW flow was noted, with all three AFW pumps running. In July of 1989, the licensee completed an engineering report on a design hydraulic calculation for the AFW system. The report indicated NPSH problems with the steam-driven (S/D) AFW pump. The report was forwarded to the Modifications and Projects Manager and technical support personnel on July 27, 1989.

On August 16, 1989, the licensee reported to the NRC that NPSH problems existed for the S/D pump running at various CST levels. The S/D pump was declared inoperable and a seven-day LCO was entered. On August 22, 1989, the licensee informed the NRC that Unit 2 was being shutdown due to NPSH problems in the AFW system when the operation of the two motor-driven (M/D) AFW pumps could not be assured.

After the licensee-initiated shutdown, an AIT was formed and arrived on site on August 28, 1989. The AIT remained on site until September 1, 1989. The inspection was documented by AIT Report Number 50-261/89-20, issued September 15, 1989. Conclusions and a brief summary are discussed below.

DISCUSSION
The AFW NPSH problem at H.B. Robinson Unit 2 was identified by the licensee's SSFI, based on calculations performed as part of the on-going design basis reconstitution during October of 1987. The Unit 2 AFW system design consisted of a CST which supplies two M/D and one S/D AFW pump by a common suction header. With all AFW pumps running at design flow conditions, and a CST level of 100%, the available NPSH would be insufficient following a main steam line break.

The licensee also identified a "friction factor" in the AFW suction piping which contributed to the AFW NPSH problem. The inner pipe wall of the AFW suction piping had experienced significant buildup of deposits and corrosion products. The buildup had increased the surface friction on the inner wall of the piping such that the friction factor was comparable to that of concrete piping. The licensee believes this condition was promoted during the 1970's and early 1980's, prior to the implementation of the more strict EPRI water chemistry guidelines.

The AIT determined that four areas of relevance contributed to this event, especially for the slow recognition of the AFW NPSH problem. These root causes are: (1) lack of initial design information for 3 pump operation in the AFW system, (2) lack of priority assigned to the licensee's SSFI findings, (3) a narrow definition of system operability, (i.e., if the pump passed the surveillance test it was declared operable, even though the AFW system may be operating in a degraded condition less than the specified flow with all pumps running), as interpreted by plant operations, and (4) plant communications proceeded at a level that did not involve plant management.

CORRECTIVE ACTIONS

The licensee has replaced the existing AFW suction piping with a larger diameter suction pipe. The new piping has 12" ID versus the previous 6" ID piping. In addition, the licensee has replaced the new piping with stainless steel. This new piping will be less susceptible to corrosion buildup.

Further corrective actions by the licensee has been to inspect and perform refurbishment on the AFW pumps. All AFW pumps were inspected for worn parts. All AFW pumps with worn parts and components were replaced or refurbished. Details of this event is described in the Notice of Violation/NRC Inspection Report Number 50-261/89-23.

Other abnormalties found since the August 1989 event were pump casing cracks in the "A" and "B" M/D AFW pumps. In addition, the "A" M/D AFW pump was found to have rotor bar cracking. The cracked rotor bars were observed at both the inboard and outboard sides of the motor. The fracture rotor bar cracks were attributed to metal fatigue due to the fact that the motor was not designed for low voltage startup capability. All pump casing cracks in the AFW pumps were ground out and repaired. The rotor bars on the "A" M/D AFW pump were subsequently replaced with a modified design.

FOLLOWUP

The NPSH problem at Robinson appears to be plant-specific and not generic. All corrective actions have been completed and the unit has returned to power operations.

John W Thompson

John Thompson
PWR Section
Events Assessment Branch

cc: C. Rossi
R. Lo
M. Reardon
H. Dance, RII

Event Followup Report 90-007

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JUN 15 1988

Docket No. 50-261
License No. DPR-23
EA 88-88

Carolina Power & Light Company
ATTN: Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering
and Construction
Post Office Box 1551
Raleigh, North Carolina 27602

Gentlemen:

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY (NRC INSPECTION REPORT NOS. 50-261/88-03 AND 50-261/88-04)

This refers to the NRC inspections conducted on January 11 - February 10, March 7, 1988, and February 11 - March 10, 1988, at the H. B. Robinson Plant. The inspections included a review of the circumstances surrounding your identification of several accident scenarios during which the minimum number of safety injection (SI) pumps necessary to meet design basis requirements would not be maintained. Those potential scenarios were identified by your staff in January and February 1988, during a review conducted in response to a letter from the NRC dated January 14, 1988. The accident scenarios involved several electrical events in which two of three SI pumps would become inoperable due to a single failure. The reports documenting these inspections were sent to you by letters dated March 14 and April 27, 1988. As a result of these inspections, failures to comply with NRC regulatory requirements were identified; and accordingly, NRC concerns relative to the inspection findings were discussed in an Enforcement Conference held on March 30, 1988. The report documenting this conference was sent to you by letter dated April 25, 1988.

The violation described in the enclosed Notice of Violation and Proposed Imposition of Civil Penalty (Notice) involved the failure of your 10 CFR Part 50, Appendix K required evaluation model to reflect the most damaging single failure relative to the ECCS safety injection (SI) subsystem. It appears that evaluations for certain single failures were not performed which resulted in the erroneous assumption that two of the three SI pumps would be operable during design basis accidents. The January/February 1988 re-evaluation conducted by you identified several electrical scenarios wherein two of the three SI pumps would become inoperable in the event of those single failures, rendering the SI function unavailable during an accident, while the evaluation model and related accident analyses described in the H. B. Robinson Updated Safety Analysis Report assumed two SI pumps required to be operable to accomplish the ECCS-SI function.

We are aware that, on the basis of your further re-evaluation of the SI system electrical design, you performed analyses after discovery of the single failure problem which indicate that only one of the three SI pumps may be needed to meet the ECCS requirements of 10 CFR 50.46. This notwithstanding, the fact remains that your earlier evaluation of the SI system failed to identify several

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single failures that would leave the plant in an unanalyzed condition with only one SI pump being operable. The NRC considers the previous plant operation with potentially only one SI pump operable rather than two pumps to be a significant reduction in the margin of safety.

To emphasize the importance of proper evaluation of ECCS system, I have been authorized, after consultation with the Director, Office of Enforcement, and the Deputy Executive Director for Regional Operations, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalty in the amount of Fifty Thousand Dollars ($50,000) for the violation described in the enclosed Notice. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," in 10 CFR Part 2, Appendix C (1988) (Enforcement Policy), the violation described in the enclosed Notice has been categorized at Severity Level III. The base value of a civil penalty for a Severity Level III violation is $50,000. The escalation and mitigation factors in the Enforcement Policy were considered and no adjustment has been deemed appropriate.

We understand that you are developing a design basis reconstitution program, the purpose of which is to verify the accuracy of the plant design basis, and that this action is being taken in view of the several design deficiencies identified during the Safety System Functional Inspection (SSFI) conducted by the NRC in April 1987. The significance of the enclosed violation and those design deficiencies identified during the SSFI serves to highlight the need for this program. and we encourage you to place priority on its timely completion.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to prevent recurrence. After reviewing your response to this Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosure will be placed in the NRC Public Document Room.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

Sincerely,

ORIGINAL SIGNED BY
J. NELSON GRACE

J. Nelson Grace
Regional Administrator

Enclosure:
Notice of Violation and Proposed
Imposition of Civil Penalty

cc w/encl:
G. P. Beatty, Jr., Vice President
Robinson Nuclear Project Department
R. E. Morgan, Plant General Manager

bcc w/encl:
NRC Resident Inspector
DRS Technical Assistant
Document Control Desk
State of South Carolina
PDR
LPDR
SECY
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JTaylor, DEDRO
JNGrace, RII
JLieberman, OE
JStefano, OE
LChandler, OGC
FIngram, PA
TMurley, NRR
Enforcement Coordinators
RI, RII, RIII, RIV, RV
BHayes, OI
SConnelly, OIA
EJordan, AEOD
EA File
ES File
DCS

B 6/2/88

OE	RII *	OGC	OE	DEDRO
JStefano	JNGrace	LChandler	JLieberman	JTaylor
5/ /88	5/ /88	5/ /88	5/ /88	6/10/88
6/10	6/10	6/10		

* RII Concurrence

OE
Publication
6/13/88

RII	RII	RII
LAReyes	GRJenkins	MLErnst
6/13/88	6/13/88	6/14/88

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Carolina Power & Light Company
H. B. Robinson Unit 2

Docket No. 50-261
License No. DPR-23
EA 88-88

During NRC inspections conducted on January 11 - February 10, March 7, and February 11 - March 10, 1988, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1988), the Nuclear Regulatory Commission proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The violation and associated civil penalty are set forth below:

10 CFR 50.46 (a) (1) requires that emergency core cooling system (ECCS) cooling performance be calculated in accordance with an acceptable evaluation model.

10 CFR Part 50, Appendix K sets forth standards for an acceptable model. Appendix K, Section D.1, "Single Failure Criterion" requires that in the accident evaluation the combination of ECCS subsystems assumed to be operative be those available after the most damaging single failure of ECCS equipment has taken place.

Contrary to the above, as of January 29, 1988, the combination of ECCS subsystems assumed to be operative in the evaluation model in the H. B. Robinson Updated Safety Analysis Report (USAR) did not reflect certain more damaging single failures of ECCS equipment, particularly the Safety Injection (SI) system. Certain single failures could have rendered two of the three SI pumps inoperable while the H. B. Robinson USAR evaluation model assumed at most one SI pump being inoperable after the most damaging single failure. The four scenarios in which the SI safety function could have been lost only leaving one SI pump operable are (1) a single failure of the sequencer relay in the safeguard sequencing logic, (2) a single failure of the emergency diesel generator (EDG) field flash circuit after loss of offsite power and loss-of-coolant conditions, (3) a single failure of the DC control power during safeguard sequencing, and (4) a single active failure in the EDG system controls.

This is a Severity Level III violation (Supplement I).

Civil Penalty - $50,000

Pursuant to the provisions of 10 CFR 2.201, Carolina Power & Light Company (licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps which will be taken to avoid further violations, and (5) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, an order may be issued to show cause why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, the licensee may pay the civil penalty by letter addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. with a check, draft, or money order payable to the Treasurer of the United States in the amount of the civil penalty proposed above or may protest imposition of the civil penalty in whole or in part by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the licensee fail to answer within the time specified, an order imposing the civil penalty will be issued. Should the licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violation listed in this Notice in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the five factors addressed Section V.B of 10 CFR Part 2, Appendix C, (1988) should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201 but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the licensee is directed to the other provisions of 10 CFR 2.205 regarding the procedure for imposing a civil penalty.

Upon failure to pay the penalty due, which has been subsequently determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282.

The responses to the Director, Office of Enforcement, noted above (Reply to a Notice of Violation, letter with payment of civil penalty, and Answer to a Notice of Violation), should be addressed to: Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region II, and a copy to the NRC Inspector at the H. B. Robinson Plant.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY:
J. NELSON GRACE

J. Nelson Grace
Regional Administrator

Dated at Atlanta, Georgia
this 15th day of June 1988

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 1 0

TITLE (4)
Loss of Safety Injection Pump Autostart Due to Eight Single-Failure Scenarios

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 1	2 8	8 8	8 8	0 0 3	0 1	1 0	2 4	8 8		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 1 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)	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)	
Don Sayre, Senior Specialist - Regulatory Compliance	8 0 3	3 8 3 - 1 2 4 2

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
B	B Q		W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

During development of a response to an NRC Request for Additional Information on the Safety Injection (SI) swing pump automatic transfer scheme, the licensee identified an original design single-failure discrepancy. Failure of the pump's DC control power supply during SI could leave only one of three SI pumps capable of automatic initiation. The licensee notified the NRC of this unanalyzed condition in accordance with 10CFR50.72(b)(1)(ii)(A) on January 28, 1988. The discrepancy was resolved and the Plant returned to full power on January 29. Later, the licensee determined that loss of a separate DC control power supply could also result in loss of emergency power for two SI pumps. The Plant was taken to cold shutdown on January 30. Further review found other single-failure scenarios, for a total of eight. Seven were resolved by February 12. The eighth was resolved on March 7 by License Amendment No. 115 for reduced power operation. On June 20, License Amendment No. 119 authorized 100 percent power operation with two SI pumps operable, each capable of automatic initiation from a separate emergency bus. This LER provides supplemental information on the event.

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1/1

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. Robinson S. E. Plant, Unit No. 2	0 5 0 0 0 2 6 1	8 8	0 0 3	0 1	0 2 OF 1 0

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

I. DESCRIPTION OF EVENT

During review of Plant documents in response to an NRC Request for Additional Information on the automatic transfer scheme for Safety Injection Pump "B" (SIP-B), the licensee identified a design discrepancy.[1,2] As originally designed, a single failure of the "B" Battery during a safety injection could result in only one SI pump (SIP-A) being available for automatic start on a Safeguards signal.[3,4,5] The tie bus between the E-1 and E-2 emergency busses would be energized from the E-1, but there would be no control power to close the breakers for SI pumps "B" and "C".[6,4] The closing power for the SIP-B breaker comes from the "B" Battery.

A special session of the Plant Nuclear Safety Committee (PNSC) was convened at 1625 hours, Thursday, January 28, 1988, to review the issue. At 1700 hours, the PNSC determined that an unanalyzed condition existed since the safety analyses for a Large Break Loss of Coolant Accident, Small Break Loss of Coolant Accident, and Main Steam Line Break assume two SI pumps available. At 1749 hours, the licensee notified the NRC Emergency Operations Center of a nonemergency one-hour reportable condition in accordance with 10CFR50.72(b)(1)(ii)(A) via the Emergency Notification System (ENS).

As initially understood, the one single failure scenario, loss of the "B" Battery, could result in the loss of the Plant's ability to automatically start two SI pumps as required by the Plant Final Safety Analysis Report (FSAR).

The condition placed the Plant into Technical Specification 3.0 which required the reactor to be in hot shutdown by 0100 hours, January 29, 1988, if the condition could not be corrected. An alternative breaker alignment and related procedure changes were pursued as an approach to eliminate the common mode failure.

At 2356 hours, January 28, a followup notification to the Emergency Operations Center was made by the licensee. In this communication, the NRC was informed that the procedure changes had been made and that a functional test of SIP-B had been performed. These actions allowed termination of the Limiting Condition for Operation at 2343 hours, January 28.

1/ Letter, K. T. Eccleston, NRC, to E. E. Utley, Carolina Power & Light Company (CP&L), H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 - REQUEST FOR ADDITIONAL INFORMATION SAFETY INJECTION PUMP B AUTO TRANSFER SCHEME, dated January 14, 1988.
2/ H. B. Robinson Unit No. 2 is a Westinghouse 700 MW Pressurized Water Reactor in commercial operation since March 1971.
3/ Battery EIIS Codes: System - EJ; Component - BTRY; Manufacturer - G185.
4/ SIP EIIS Codes: System - BQ; Component - P; Manufacturer - W318.
5/ Safeguards EIIS Codes: System - JE; Component - Not Available; Manufacturer - W120.
6/ Bus EIIS Codes: System - EK; Component - BU; Manufacturer - W120.

Technical Specification Action Statement 3.0 when entered required hot shutdown in eight hours. The Plant had begun a 10 percent per hour descent in power. Prior to hot shutdown, however, the breaker arrangement problem was resolved and the Plant was returned to full power at 0535 hours, January 29.

Later in the day, January 29, during follow-up of the event, it was discovered that a single failure of the "A" Battery could result in a loss of the "A" Emergency Diesel Generator during a design basis event since the "A" Battery supplies control power to this diesel generator. Loss of the "A" Emergency Diesel Generator (and emergency bus E-1) would result in the loss of SIP-A and SIP-B since the tie bus normal feed breaker from E-1 would also be lost due to the assumed failure of the "A" Battery. Since the normal tie bus feeder breaker would not automatically open, the interlock necessary for the alternate supply breaker from E-2 to close would not be satisfied. Therefore, without manual actions, SIP-B would not start. This again placed the Plant in an unanalyzed condition. Technical Specification 3.0 was entered, requiring the reactor to be in hot shtudown in eight hours and cold shutdown in the next 30 hours. At 1410 hours, the licensee notified the Emergency Operations Center of this unanalyzed condition in accordance with 10CFR50.72(b)(1)(ii)(A) via the ENS. Since it appeared that other single failures could be postulated, the licensee decided to conduct a more detailed review. The Plant proceeded to hot shutdown, then to cold shutdown.

At 2036 hours, January 29, the licensee made a followup notification to the Emergency Operations Center to report the reactor in hot shutdown at 2026 hours.

At 2035 hours, January 30, the licensee made a followup notification to the Emergency Operations Center to report the reactor in cold shutdown at 1942 hours.

The Plant entered a forced outage for resolution of the conditions identified and to allow for further design review, to determine whether there may be other single-failure scenarios. This continued investigation identified a total of eight scenarios under which the electrical distribution system may be outside of the analysis for single-failure vulnerability. The Plant remained in cold shutdown pending resolution of the concerns.

Seven of the eight scenarios were resolved by the licensee by February 12, 1988. Resolution of the remaining scenario required additional extensive engineering review and was addressed on an interim basis by analysis justifying the need for only one SI pump at steady state reactor core power levels no greater than 60 percent (1380 Megawatts thermal). A request for a license amendment to address restricted power operation was submitted to the NRC on February 24, 1988.[7]

7/ Letter, M. A. McDuffie, CP&L, to NRC, Serial: NLS-88-044, dated February 24, 1988.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. Robinson S. E. Plant, Unit No. 2	0 5 0 0 0 2 6 1	8 8	— 0 0 3 —	0 1	0 4 OF 1 0

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

The eight single-failure scenarios have been described in letters submitted to the NRC.[7,8]

See Section VI.C.

II. CAUSE OF EVENT

The cause of the single-failure susceptibility appears to be inherent in the design of SIP-B and the emergency AC and DC distribution systems in how they provide control power and motor power for SIP-B. Specifically, the SIP-B was designed to be powered automatically from either the "A" or "B" Train (480V emergency power) via a tie bus arrangement (Figure 1). Power would be preferentially supplied by the "A" Train (bus E-1) through a tie breaker. If the "A" Train power was unavailable, the selection logic would sense this tie breaker open and the opposite tie breaker would be closed by the SI sequencer, providing power from the "B" Train (bus E-2). However, control power for SIP-B is provided by only the "B" Train ("B" DC distribution system). It was this configuration (two trains of power, one train of control) and the interrelation of the "A" and "B" Train logics associated with automatic starting of SIP-B that created the various combinations of single-failure scenarios.

The design deficiency occurred during the original design of the Plant and details as to the reasons have been investigated. At the time of original design, the active failure assumptions were less conservative than today.

See Section VI.C.

III. ANALYSIS OF EVENT

The single failure resulting in the potential loss of two of the three automatically initiated SI pumps resulted in an unanalyzed condition since the safety analyses assumed a flow from two SI pumps to mitigate the consequences of the accidents analyzed. As the first single-failure susceptibility was recognized, immediate corrective action was taken to change breaker alignment. However, a second aspect was recognized shortly thereafter and it was recognized that a more indepth review was needed to determine the potential for additional single failures. Accordingly, the reactor was taken to cold shtudown.

Analyses were conducted to support return to power operation. Results from these analyses have been used to provide a more detailed event analysis.

See Section VI.C.

8/ Letter, M. A. McDuffie, CP&L, to NRC, Serial: NLS-88-035, dated February 12, 1988.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. Robinson S. E. Plant, Unit No. 2	0 5 0 0 0 2 6 1	8 8	— 0 0 3	— 0 1	0 5 OF 1 0

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

IV. CORRECTIVE ACTION

Corrective action for each of the scenarios identified are detailed in the previously referenced correspondence. 7,8 Permanent corrective action for one scenario required more extensive engineering reviews. Accordingly, as an interim measure to return the Plant to operation, analyses were performed to establish a power level at which operation with only two available automatically initiated SI pumps (and assuming a single failure of one) could be justified. That power level was determined to be 60% of rated power (1380 Megawatts thermal). Accordingly, a modification was implemented to remove the automatic start feature of SIP-B and auto closure of the bus tie breakers.[9] As a longer term solution and as additional corrective actions were implemented, appropriate licensing action was initiated.

See Section VI.C.

V. ADDITIONAL INFORMATION

A. Failed Component Identification

The emergency electrical distribution DC system is of Westinghouse design, 125 volts, two independent battery banks with separate battery chargers fed by the two emergency diesel generators.

B. Previous Similar Events

No other postulated single-failure scenarios have been identified or reported on with regard to the SI emergency electrical DC power distribution system.

LER-87-026-00 of November 29, 1987, reported a potential for degraded recirculation flow for the Residual Heat Removal Pumps due to a common miniflow recirculation configuration.[10]

LER-87-030-00 of December 17, 1987, reported a potential single failure that could prevent two redundant Safety Injection and Residual Heat Removal Valves from opening remotely from the Unit 2 Control Room.[11]

LER-88-003-00 of February 27, 1988, provided the original report on this event.[12]

9/ Plant Modification M-947, SI PUMP AVAILABILITY UPGRADE.
10/ Letter, R. E. Morgan, CP&L, to NRC, Serial: RNPD/87-5785, dated November 29, 1987.
11/ Letter, R. E. Morgan, CP&L, to NRC, Serial: RNPD/87-5941, dated December 17, 1987.
12/ Letter, R. E. Morgan, CP&L, to NRC, Serial: RNPD/88-1084, dated February 27, 1988.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. Robinson S. E. Plant, Unit No. 2	0 5 0 0 0 2 6 1	8 8	— 0 0 3	— 0 1	0 6 OF 1 0

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

C. Supplemental Information

The NRC conducted a routine, announced inspection from January 11 through February 10, and March 7, 1988, including an onsite followup of this event.[13]

The licensee and NRC held a meeting on February 10, 1988 to discuss the proposed modification of the onsite emergency electrical distribution system to correct the following design deficiencies resulting in single-failure vulnerability of the system under certain conditions:[14]

1. E-1/E-2 bus tie breaker misalignment
2. Train "A" safeguards sequence interlock relay with Train "B" safeguards sequencer.
3. Postulated break in internal wiring in safeguards sequencers.
4. Loss of Emergency Diesel Generator field flash circuitry during Loss of Offsite Power SI conditions.
5. Loss of DC control power to E-1/E-2 emergency busses.

The NRC provided a Confirmation of Action letter on the NRC's understanding of commitments made during the February 10, 1988 meeting.[15] The licensee responded with commitments to resolve the concerns regarding SI System operability.[8] This response included the design basis for equipment modification, single-failure scenarios and corrective actions, acceptance testing, and a training schedule.

The NRC conducted a special, announced inspection on February 12 and 13, 1988 to observe post-modification testing to verify the Plant's ability to automatically start two SI pumps after each of the five postulated single failure events.[16,17]

13/ Letter, J. N. Grace, NRC, to E. E. Utley, CP&L, NRC INSPECTION REPORT NO. 50-261/88-03, dated March 14, 1988.
14/ Letter, R. H. Lo, NRC, to CP&L, MEETING SUMMARY FOR FEBRUARY 10, 1988 MEETING ON MODIFICATIONS OF EMERGENCY ELECTRICAL DISTRIBUTION SYSTEM, H. B. ROBINSON UNIT NO. 2, dated February 23, 1988.
15/ Letter, J. N. Grace, NRC, to E. E. Utley, CP&L, CONFIRMATION OF ACTION LETTER, dated February 11, 1988.
16/ Letter, A. R. Herdt, NRC, to E. E. Utley, CP&L, NRC INSPECTION REPORT NO. 50-261/88-05, dated March 9, 1988.
17/ Plant Special Procedure No. 796, VERIFICATION OF SAFETY INJECTION PUMP AVAILABILITY AND SAFEGUARDS SEQUENCE FUNCTIONS.

*U.S. GPO 1988-0-824-538/455

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. Robinson S. E. Plant, Unit No. 2	0 5 0 0 0 2 6 1	8 8	0 0 3	0 1	0 7 OF 1 0

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

The licensee and the NRC held a meeting on February 16, 1988 to discuss the Loss of Coolant Accident (LOCA) analyses for safety injection with a single failure.[18] This meeting included presentation of a justification for startup and operation of the Plant with 15 x 15 fuel in conformance with the accident criteria of 10CFR50.46.

On February 24, 1988, the licensee submitted an emergency request for a license amendment concerning the SI System.[7]

The SIP-B autostart capability was deleted by a separate Modification. This action corrected single failure susceptibilities that could result in abnormal voltages frequency which could cause damage to the two SIP motors connected to the same emergency bus. The Modification changed the breaker logic feeding SIP-B by providing manual control of the pump versus an automatic control scheme.[19]

On February 26, 1988 the licensee submitted a supplement to the February 24 emergency request for a license amendment.[20]

The licensee directed an onsite investigation into the SIP-B concerns.[21] A separate evaluation by the licensee Nuclear Fuel Section of the effect of an increase of 10 seconds in the response of one SIP due to a malfunction in the emergency power circuit that disables SIP-A and SIP-C and delays the starting of SIP-B. The conclusion was insignificant on the calculated consequences of the accident.

On March 1, 1988 the licensee submitted a second supplement to the February 24 emergency request for a license amendment.[22]

On March 2, 1988 the facility Nuclear Steam Supply System designer provided a letter to the licensee indicating that the facility and at least four other Plants of similar vintage were originally designed to require only one of two SI pumps to be online to satisfy minimum safeguards flow requirements.[23] Three SI pumps were incorporated into the original designs, however, with the third pump considered an installed spare. Subsequently, the designer determined that additional safeguards flow beyond that of a single pump was needed for

18/ Letter, R. H. Lo, NRC, to CP&L, SUMMARY OF FEBRUARY 16, 1988 MEETING ON LOSS OF COOLANT ACCIDENT (LOCA) ANALYSIS FOR SAFETY INJECTION WITH SINGLE FAILURE, H. B. ROBINSON, UNIT NO. 2, dated February 28, 1988.

19/ Plant Modification M-951, SI PUMP "B" DELETION OF AUTOSTART.

20/ Letter, M. A. McDuffie, CP&L, to NRC, Serial: NLS-88-052, dated February 26, 1988.

21/ Plant Operating Experience Report No. 88-05, SI PUMP "B" INVESTIGATION, FEBRUARY 1988.

22/ Letter, L. W. Eury, CP&L, to NRC, Serial: NLS-88-057, dated March 1, 1988.

23/ Letter, G. O. Percival, Westinghouse, to R. E. Morgan, CP&L, CAROLINA POWER & LIGHT COMPANY H. B. ROBINSON UNIT 2 SAFETY INJECTION ELECTRICAL DESIGN, Serial: CPL-88-515, dated March 2, 1988.

conservatism in mitigating a steam line break accident and provide a faster change in reactivity. As a result, the most economical solution at the time was to automatically start the spare pump versus changing the pump/fluid system design. The concept of a swing pump, capable of being automatically powered from either Train was devised. This design was consistent with the single-failure criteria and philosophy of the time although no longer acceptable in light of current technical knowledge.

The NRC issued Plant Operating License Amendment No. 115 on March 7, 1988, restricting operation of the Plant below 1380 MegaWatts thermal with two SI pumps operable to mitigate the consequences of a Loss of Coolant Accident.[24,25]

The NRC provided a Confirmation of Concurrence letter on March 8, 1988 which detailed the licensee commitments made at the February 10, 1988 meeting and concurred with Plant restart.[26]

On March 15, 1988 the licensee provided a letter on the SIP-B autotransfer scheme.[27]

The licensee and the NRC held a meeting on March 30, 1988, to discuss the findings of NRC Inspection Report No. 50-261/88-03.[28,29]

On May 7, 1988 the licensee requested a license amendment to remove the operating restrictions of Amendment No. 115 and permission to return to 100 percent reactor power.[30]

The licensee provided the NRC an analysis on May 9, 1988 which was approved by the NRC on June 20, 1988.[31]

24/ Telephone Conference Call, Lainas/Adensom/Lo/Loflin, AUTHORIZATION OF PROPOSED TECHNICAL SPECIFICATIONS CHANGES TO ALLOW PLANT RESTART, dated March 7, 1988.

25/ Letter, R. H. Lo, NRC, to E. E. Utley, CP&L, ISSUANCE OF AMENDMENT NO. 115 TO FACILITY OPERATING LICENSE NO. DPR-23 REGARDING OPERATION OF PLANT BELOW 1380 MWt, dated March 7, 1988.

26/ Letter, J. N. Grace, NRC, to E. E. Utley, CP&L, CONFIRMATION OF CONCURRENCE, dated March 8, 1988.

27/ Letter, L. I. Loflin, CP&L, to USNRC, Serial: NLS-88-065, dated March 15, 1988

28/ Letter, J. N. Grace, NRC, to E. E. Utley, CP&L, CONFIRMATION OF ENFORCEMENT CONFERENCE, H. B. ROBINSON DOCKET NO. 50-261, dated March 17, 1988.

29/ Letter, J. N. Grace, NRC, to E. E. Utley, CP&L, ENFORCEMENT CONFERENCE SUMMARY (NRC INSPECTION REPORT NO. 50-261/88-03), dated April 25, 1988.

30/ Letter, M. A. McDuffie, CP&L, to NRC, Serial: NLS-88-111, dated May 7, 1988.

31/ Letter, R. H. Lo, NRC, to E. E. Utley, CP&L, ISSUANCE OF AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. DPR-23, dated June 20, 1988.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. Robinson S. E. Plant, Unit No. 2	0 5 0 0 0 2 6 1	8 8	— 0 0 3	— 0 1	0 9 OF 1 0

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

On May 16, 1988 the licensee submitted corrected information for the submittal of May 7.[32]
On May 20, 1988 the licensee submitted corrected information for the submittal of May 7.[33]

On June 15, 1988 the NRC issued a Notice of Violation to the licensee regarding the single-failure concerns.[34]

The NRC issued Plant Operating Licensee Amendment No. 119 on June 20, 1988 allowing Plant operation at a steady state reactor core power level not in excess of 2300 MegaWatts thermal with two SI pumps operable, each capable of automatic initiation from a separate emergency bus.[31]

On July 15, 1988, the licensee responded to the Notice of Violation.[35]

32/ Letter, L. I. Loflin, CP&L, to NRC, Serial: NLS-88-127, dated May 16, 1988.
33/ Letter, L. I. Loflin, CP&L, to NRC, Serial: NLS-88-129, dated May 20, 1988.
34/ Letter, J. N. Grace, NRC, to E. E. Utley, CP&L, NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY, dated June 15, 1988.
35/ Letter, L. W. Eury, CP&L, to NRC, Serial: NLS-88-152, dated July 15, 1988.

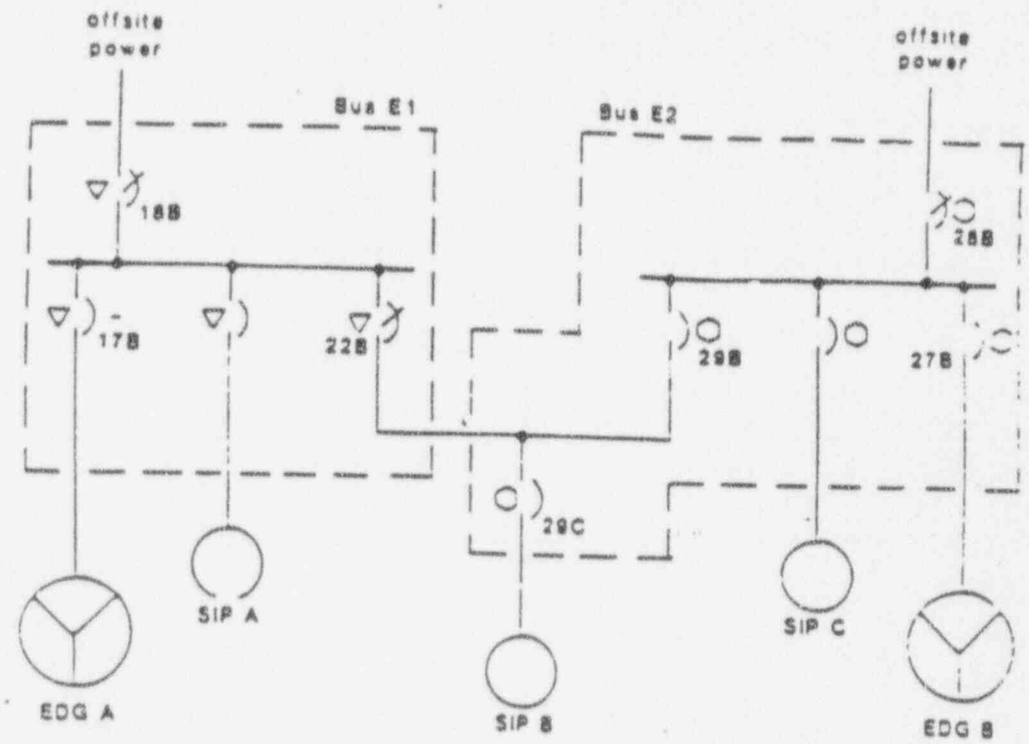
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. Robinson S. E. Plant, Unit No.2	0 5 0 0 0 2 6 1	8 8	0 0 3	0 1	1 0 OF 1 0

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

Figure I

Normal Emergency Bus Lineup

(Before Jan 28, 1988)



▽ = Breaker control from Train A battery

○ = Breaker control from Train B battery

Breaker - open

Breaker - closed

EDG - Emergency Diesel Generator

SIP - Safety Injection Pump

CP&L

Carolina Power & Light Company

ROBINSON NUCLEAR PROJECT DEPARTMENT
POST OFFICE BOX 790
HARTSVILLE, SOUTH CAROLINA 29550
OCT 24 1988

Robinson File No: 13510C

Serial: RNPD/88-3511
(10 CFR 50.73)

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
LICENSEE EVENT REPORT 88-003-01

Gentlemen:

The enclosed Supplemental Licensee Event Report (LER) is submitted in accordance with 10 CFR 50.73 and NUREG-1022 including Supplements No. 1 and 2. This submittal should replace existing copies of the original report of February 27, 1988.

Very truly yours,

R E Morgan

R. E. Morgan
General Manager
H. B. Robinson S. E. Plant

Enclosure

cc: Dr. J. N. Grace
Mr. L. W. Garner
INPO

IE22
1/1

EVENT FOLLOWUP REPORT 87-177
50.72 EVENT #10849 DECEMBER 2, 1987
PLANT- H.B.ROBINSON UNIT 2
SUBJECT: UNANALYZED ECCS FAILURE MODE
PROJECT MANAGER-KENNETH ECCLESTON
COGNIZANT ENGINEER-WALTON JENSEN

PROBLEM

Loss of one vital bus would prevent opening of safety injection valves SIS 863A and 863 B

CAUSE

The preventive interlocks for both valves have a common power source.

SAFETY SIGNIFICANCE

At least one of the valves must be opened to establish high pressure safety injection in the recirculation period following a small break LOCA to provide core cooling.

DISCUSSION

On December 2, 1987 the licensee reported a design deficiency in the safety injection system of H.B. Robinson Unit 2. The deficiency was identified by Westinghouse in a letter dated November 3, 1987 which described a similar problem at Turkey Point and suggested that Carolina Power and Light review the interlock logic and power arrangements for the valves at H. B. Robinson. The H. B. Robinson valves are equipped with interlocks to prevent their opening when the RHR pump discharge pressure is above approximately 200 psig. This condition occurs when the reactor is in Mode 4. If the valves were open in Mode 4 a direct path would exist for primary coolant to be lost to the refueling water storage tank (see the attached figure). Either high RHR pump discharge pressure or loss of power to the single vital bus supplying the interlocks for both valves would prevent the valves from being opened.

Following a loss of coolant accident the RHR pumps, the high pressure safety injection pumps and the containment spray pumps initially all take suction from the refueling water storage tank (RWST). When the low RWST level alarm occurs, reactor operators are instructed to switch suction for the RHR pumps from the RWST to the containment sump. The low RWST level alarm would occur after about 20 minutes following a large break LOCA and after about one hour following a small break LOCA. Although the RHR pumps are stopped during this process the high pressure SI pumps would continue to inject. When the RHR pumps are realigned the high pressure SI pumps are stopped and aligned to take suction from the RHR pumps' discharge. For large break LOCAs either the RHR pumps or the high pressure SI pumps would provide a continuous flow of water to the core during the switchover process. For small break LOCAs ECCS flow would be interrupted while the high pressure SI pumps were stopped. This is because the RHR pumps cannot inject into the reactor system under the elevated pressure conditions that would exist following a small break LOCA. Westinghouse has calculated that for a typical plant, ECCS could be interrupted for at least 10 minutes during the switchover following a small break LOCA with acceptable results.

In the early phase of the post LOCA procedure the operators are instructed to restore power to the valve operators of SIS 863 A and SIS 863 B. This is because the technical specifications require electric power to be locked out from the valves when the reactor is at power. At least one hour would be available to restore power following a small break LOCA. Even with power restored the valves would not open if vital instrument bus #4 which powers both interlocks were to fail. If the valves could not be opened high pressure safety injection would be lost during the recirculation period. For large breaks the RHR pumps would inject sump water directly into the reactor system but could not inject for high reactor system pressures typical of small break LOCAs. The licensee's temporary solution is to defeat the interlock with jumpers so that the valves can be opened or closed from the control room at any time when power to the motor operators is not locked out. The interlock will be restored when the reactor is shutdown by removing the jumper wires.

GENERIC IMPLICATIONS
A similar problem was identified at Turkey Point in July 1984 and corrected by providing redundant vital power to the interlock in 1985. Turkey Point filed a Part 21 notification (attached). The problem was entered in the INPO Notepad. As discussed in the attached memorandum from C. Rossi, March 8, 1988 no additional generic communication is warranted at this time.

FOLLOWUP
The Reactor Systems Branch is evaluating operator action for manual ECCS switchover under TAC 66653. This issue should be included in TAC 66653.

STATUS
EAB is continuing to follow the RSB generic review of ECCS switchover.

Walton Jensen

Walton Jensen
PWR Section
Events Assessment Branch

cc: K. Eccleston
C. Rossi
W. Hodges
R. Jones