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SAYRE,D. Carolina Power & Light Co.
MORGAN,R.E. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 88-003-01:on 880128,loss of safety injection pump autostart due to eight single-failure scenarios.
W/8 ltr.

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
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EXPIRES: 8/31/88

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

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
N		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
1 0 0		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)	
	X	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
	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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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.

*U.S.GPO:1986-0-624-538/455

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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.

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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.

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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.

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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.

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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.

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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.

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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.

★U.S.GPO:1986-0-624-638/455

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

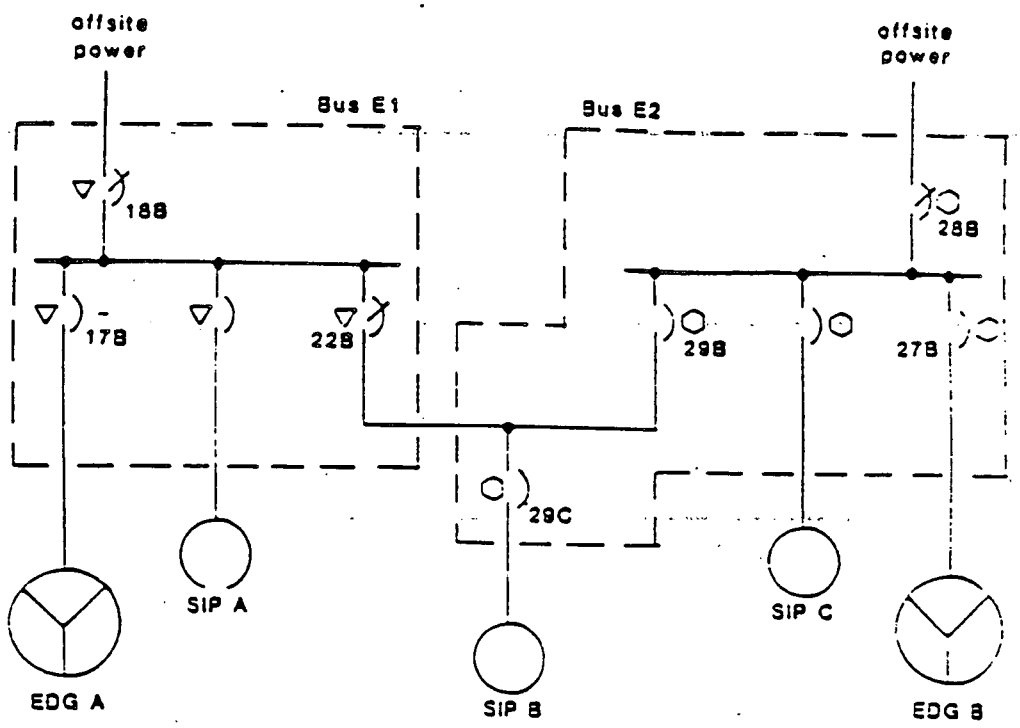
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
H. B. Robinson S. E. Plant, Unit No.2	0 5 0 0 0 2 6 1	8 8	— 0 0 3	— 0 1	1 0 OF 1 0

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

*U.S.GPO:1988-0-624-538/456

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

Enclosure

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

IE22
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