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ACCESSION NBR: 8710200569 DOC. DATE: 87/10/15 NOTARIZED: NO DOCKET #
FACIL: 50-400 Shearon Harris Nuclear Power Plant, Unit 1, Carolina 05000400

AUTH. NAME	AUTHOR AFFILIATION
WALLACE, M.	Carolina Power & Light Co.
WATSON, R. A.	Carolina Power & Light Co.
RECIP. NAME	RECIPIENT AFFILIATION

SUBJECT: LER 87-054-00: on 870915, determined that more limiting single failure might exist & on 870916 two addl potential failure modes identified & plant shutdown. Mod made to solid state protection sys. W/871015 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 9
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: Application for permit renewal filed. 05000400

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	REG FILE 02	1	1	RES DEPY GI	1	1
	~~RES TELFORD, J~~	1	1	RES/DE/EIB	1	1
	RGN2 FILE 01	1	1			
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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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)
SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1	0 5 0 0 0 4 0 0	1 OF 0 8

TITLE (4) UNANALYZED CONDITION - ISOLATION OF AUXILIARY FEEDWATER CAUSED BY LOSS OF VITAL DC BUS 1B-SB COINCIDENT WITH LOSS OF OFF-SITE POWER

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 9	1 5	8 7	8 7	0 5 4	0 0	1 0	1 5	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
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	
M. WALLACE - REGULATORY COMPLIANCE	9 1 9	3 6 2 - 2 7 1 9

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

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

ABSTRACT

On September 15, 1987, at 2000, with the plant in Mode 1 at 100% power, it was determined that a more limiting single failure than had been previously analyzed might exist, and that if valid, continued operation of the plant was not justified. A plant shutdown commenced at 2015, and the unit was off-line at 0040 on September 16, 1987. The unanalyzed failure involved a loss of 'B' vital DC bus causing a failure of the turbine-driven Auxiliary Feedwater (AFW) pump and the 'B' motor-driven AFW pump. The Final Safety Analysis Report (FSAR) accident analysis for Main Feedwater Line Break assumes the availability of both motor-driven pumps 'A' and 'B'. The plant was cooled down to Mode 4 at 0440 on September 16, 1987, where Auxiliary Feedwater is not required to be operable by Technical Specifications.

On September 16, 1987, two additional potential failure modes were identified; one involved the spurious failure of a relay in the Solid State Protection System (SSPS) causing inadvertent isolation of AFW to one steam generator. The second was that failure of 'B' vital DC bus coincident with a loss of off-site power would isolate AFW to all three steam generators.

Reanalysis of accidents with a reduced AFW capability was done, and it was verified that design requirements are met, assuming first one motor-driven pump is available, and second a single steam generator is inadvertently isolated. A modification was made to the SSPS, which changed one channel of AFW isolation relays and bistables from "deenergize to actuate" to "energize to actuate", thus eliminating the single failure concern which caused complete isolation of AFW. Upon completion of these analyses and modifications, the unit was returned to service.

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NRC Form 366 (9-83)

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

DESCRIPTION:

On September 15, 1987, at 2000, with the plant in Mode 1 at 100% power, it was determined that a more limiting single failure than had been previously analyzed might exist, and that if valid, continued operation of the plant was not justified. A plant shutdown commenced at 2015, and the unit was off-line at 0040 on September 16, 1987. The unanalyzed failure involved a loss of vital DC bus 'B' (EIIS:EJ) causing a failure of the turbine-driven Auxiliary Feedwater (AFW) pump (EIIS:BA) and the 'B' motor-driven AFW pump (EIIS:BA). The Final Safety Analysis Report (FSAR) accident analysis for Main Feedwater Line Break assumed the availability of both motor-driven pumps 'A' and 'B' (EIIS:BA). The 'B' vital DC bus was operable and energized at the time of discovery. The plant was cooled down to Mode 4 at 0440 on September 16, 1987, where Auxiliary Feedwater (EIIS:BA) is not required to be operable by Technical Specifications.

The above defect led to consideration of three new accident scenarios. The new scenarios are as follows:

Scenario I: The loss of vital DC bus 1B-SB coincident with a loss of off-site power causes loss of one motor-driven AFW pump and the turbine-driven AFW pump.

The DC failure would disable the 'B' emergency diesel generator (EDG) (EIIS:EK) starting capability and tie breaker control power, resulting in a failure to energize vital AC, if the DC failure occurs prior to loss of off-site power. The loss of DC power also would cause the failure of the TDAFW governor control (EIIS:BA) and, if loss of DC occurs before load sequencing is complete, the loss of 'B' AFW breaker (EIIS:BA) closing capability.

The FSAR description of the AFW system (FSAR Section 10.4.9) stated that a single motor driven AFW pump would have sufficient flow to meet the flow demand of the Chapter 15.0 analysis. Control systems were relied upon to isolate AFW flow to a faulted steam generator and isolate a motor driven AFW pump recirculation line if the other motor driven pump failed. However, the FSAR Chapter 15.0 analysis did not explicitly model the isolation of AFW flow to the faulted steam generator and instead used the argument that a single failure would result in at least two motor driven pumps (900 gpm total) being available. A review of the FSAR Chapter 15.0 analysis could not verify that the AFW isolation would be timely.

Scenario II: The inadvertent actuation of a relay causes isolation of AFW flow to one intact steam generator (EIIS:SB). This single failure was not analyzed in Chapter 15 of the FSAR.

Scenario III: The loss of vital DC bus 1B-SB coincident with a loss of off-site power allowed the Engineered Safety Features Actuation System (EIIS:JE) to isolate AFW from all three steam generators.

The instruments providing inputs to the Engineered Safety Features Actuation System are powered from safety related inverters. (SI, SII, SIII or SIV).

As shown in FSAR Figure 8.1.3-3 (Attachment 1), DC bus 1B-SB is the alternate power supply to the inverters supplying instrument busses SII and SIV. The safety-related AC power (EIIS:EL) is the normal supply to these inverters through train B safety-related, diesel generator connectable AC busses (EIIS:EK). Upon a loss of DC bus 1B-SB, the instrument

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

DESCRIPTION: (continued)

busses remain powered via the AC input to the inverters. However, this power supply is deenergized upon a postulated loss of off-site power. Emergency power is supplied by the 1B-SB emergency diesel generator; however, it requires DC bus 1B-SB to power the starting air valves (EIIS:EK), field flashing and breaker control power. Since the DC bus is also postulated to fail, the diesel generator will not be available, and the two instrument busses will deenergize.

Instrument bus SII is the power supply for Process Instrument Cabinet (EIIS:EF) (PIC) 2, and bus SIV is the power supply for PIC 4. In addition, these busses provide power to the Solid State Protection System (SSPS) train B (EIIS:JG). Upon a loss of power to the PICs, the instrument bistables deenergize to a "fail-safe" position. As shown in FSAR Figure 7.1.1-1 (Attachment 2), the output of these bistables, referred to as Channel II and Channel IV, is input into both trains of solid state logic.

Shearon Harris Technical Specification 3.3.2 requires the capability to automatically isolate AFW to a faulted steam generator. As shown in FSAR Figure 7.2.1-1 (Attachment 3), this requirement is met by comparing steam generator pressures from instrument channels 2, 3, and 4 using a two out of three logic. The failure of any one channel will not cause inadvertent actuation, and since the bistables deenergize to actuate, the failure of any one channel will not prevent actuation when required. Since the postulated failure of DC bus 1B-SB causes a loss of both channels 2 and 4, the actuation logic for AFW isolation will be made up and AFW isolation will occur for all three steam generators. The two deenergized PICs provide trip signals to the Train A logic which will respond to the input. This is sufficient to complete the required 2/3 logic even if the valid Channel III input remains untripped.

As discussed in Scenario I, the failure of DC bus 1B-SB causes the turbine driven AFW pump and "B" Motor Driven AFW pump to fail. This leaves only the 1A-SA motor driven AFW pump. With only the 1A-SA AFW pump operating, an interlock required by Technical Specification 3.7.1.2 closes the valve in the pump recirculation line to increase the flow available to the steam generators. With the flow path to all three Steam Generators blocked there is no remaining flowpath to ensure a minimum flow and protect the 1A-SA pump.

Reanalysis of Scenario I & II were performed with a reduced AFW capability. It was verified that design requirements and FSAR acceptance criteria are met assuming only 430 gpm (within the capacity of one motor-driven pump) is available, or if a single steam generator is inadvertently isolated. These analyses were performed using the same analytical methods used for the previous FSAR analysis. The analysis, however, relied on ANSI/ANS-5.1-1979 "Decay Heat Power In Light Water Reactors" vice the 1971 version used in the previous FSAR analysis. This is considered acceptable by CP&L because the NRC had approved the use of this decay heat curve for similar analyses for other licensees.

To prevent the inadvertent isolation of AFW to all three steam generators, a modification was made to the SSPS and PIC 4. The modification changed one channel of AFW isolation relays (EIIS:BA) and bistables (EIIS:BA) from "deenergize to actuate" to "energize to actuate". Thus if the single failure deenergized both channels II and IV, the correct action of channel III would determine if an AFW isolation signal would result.

DESCRIPTION: (continued)

Upon completion of these analyses and modifications on September 24, 1987, the unit was returned to service. The plant was out of service for 10 days. The plant was maintained in Mode 4 or Mode 5 where AFW is not required by Technical Specifications for Reactor Coolant System (RCS) heat sink. This event was reported to the NRC per 10CFR21 in a letter dated September 22, 1987.

CAUSE:

The cause of this event was a design error. The plant design was a combined engineering effort between Ebasco, Westinghouse, and CP&L.

The impact of a DC bus failure occurring prior to or coincident with a loss of off-site AC power was not specifically analyzed, or justified in the FSAR as not requiring consideration. Further review of requirements to consider a DC bus failure coincident with a loss of offsite power in the licensing basis for the plant has indicated that such a failure need not be analyzed. This review considered the applicable GDC's, Regulatory Guides, and IEEE Standards. The review concluded that the design did not have to consider a single failure during the transition between onsite and offsite power.

CP&L has, however, corrected the plant design to mitigate such a failure as described below.

CP&L has no indication that the AFW isolation logic was specifically reviewed for the affects of inadvertent actuation of SSPS output relays.

ANALYSIS:

This event is being reported in accordance with 10CFR50.73(a)(2)(ii)(A) due to an unanalyzed condition.

The failure mode, mechanism, and effects are described in the event description.

The complete failure of a DC bus is a very low probability event, since the bus has two safety related chargers and a battery bank available to maintain it energized. Technical Specifications allow only two hours of operation with a deenergized DC bus before requiring a unit shutdown. If a loss of off-site power had also occurred simultaneous with or following the DC bus failure, the Auxiliary Feedwater (AFW) lineup would have been: turbine-driven (TD) AFW pump inoperable, 'B' motor-driven (MD) AFW pump inoperable; 'A' MDAFW pump operating; all flow control valves (FCVs) (EIIS:BA) in the MDAFW pump header closed; all steam generator isolation valves in the TDAFW flow paths closed; recirculation valve (EIIS:BA) for 'A' MDAFW pump closed. The operator could take action to open the FCVs and establish AFW, but probably not in time to prevent damaging the 'A' AFW pump due to its operation at shutoff head with no recirculation path.

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

ANALYSIS: (continued)

A "bleed and feed" operation of the Reactor Coolant System (RCS) (EIIS:AB) would then be initiated in accordance with Emergency Operating Procedures. "Bleed and feed" requires opening the Pressurizer PORVs (EIIS:AB). The actuation of the PORVs is independent of the "B" DC power. However, two of the three PORV block valves are powered from the "B" AC power. If these valves are closed as permitted by Technical Specifications, they would be unavailable. This situation is analyzed as part of the Westinghouse Owner's Group Emergency Response Guidelines, (ERGs) Background Volume. The ERGs state that two PORVs are required to assure successful bleed and feed. Action to restore 'B' AC power and/or AFW capability would therefore be needed to stabilize the reactor and establish an adequate heat sink.

Engineering reviews discovered each failure mechanism. The requirement by regulations to consider a DC bus failure coincident with loss of off-site power was not clear.

CORRECTIVE ACTIONS:

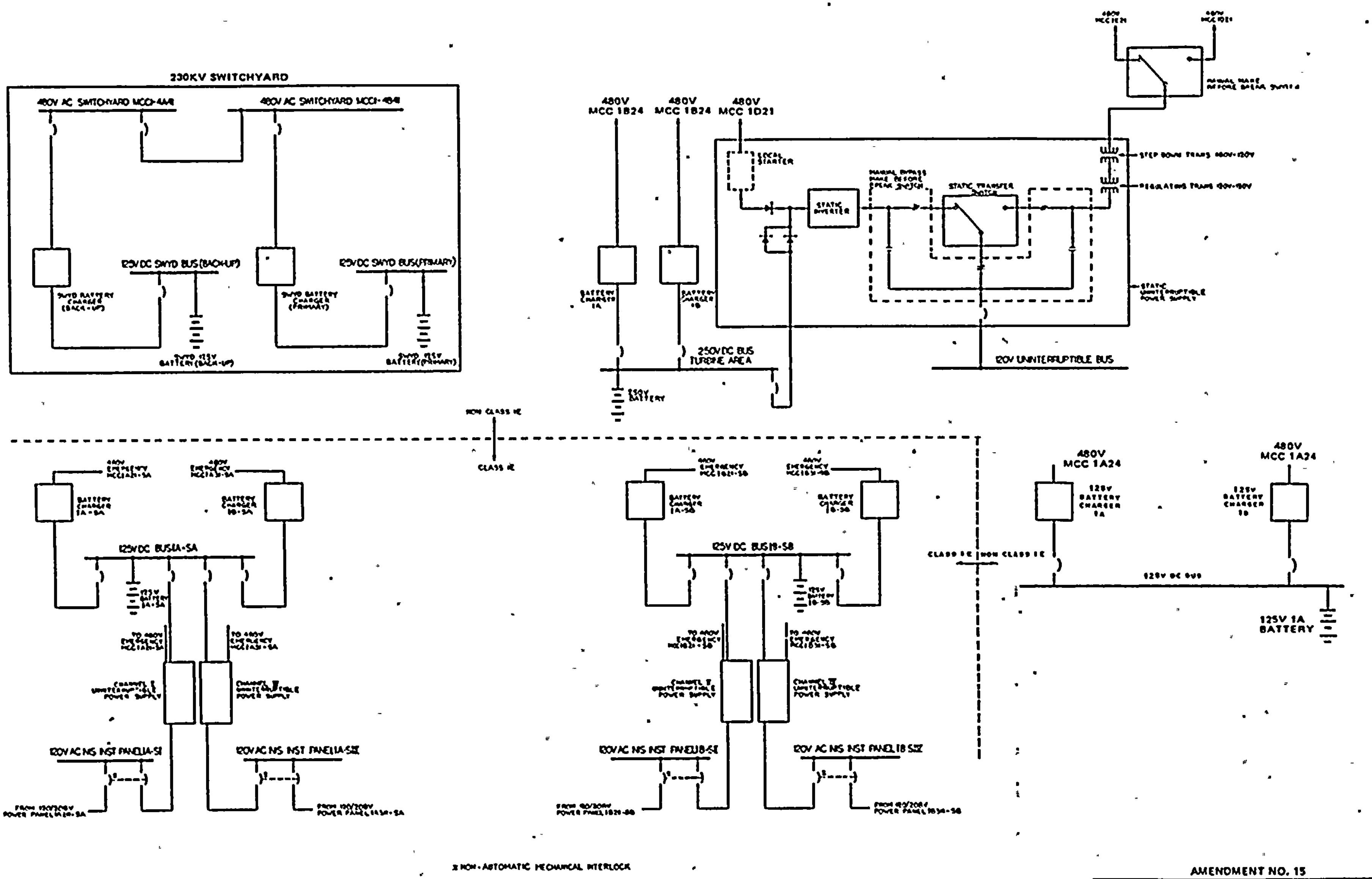
1. A reanalysis of FSAR accidents was satisfactorily completed assuming only one motor driven Auxiliary Feedwater (AFW) pump available. (used 1979 ANS5.1 Decay Heat Model)

2. The Solid State Protection System (SSPS) and Process Instrumentation Cabinet 4 were modified to change channel IV input for Steam Generator differential pressure - AFW isolation from "deenergize to actuate" to "energize to actuate." With a 2/3 logic, this results, for the 'B' Vital DC bus failure, in one channel tripped, one not tripped, and the third channel available to actuate if required.

3. The SSPS was reviewed to determine if failure of a train of DC power would create additional adverse consequences. No additional problems were found in this review.

4. A 10CFR21 notification was made on September 22, 1987.

ATTACHMENT 1



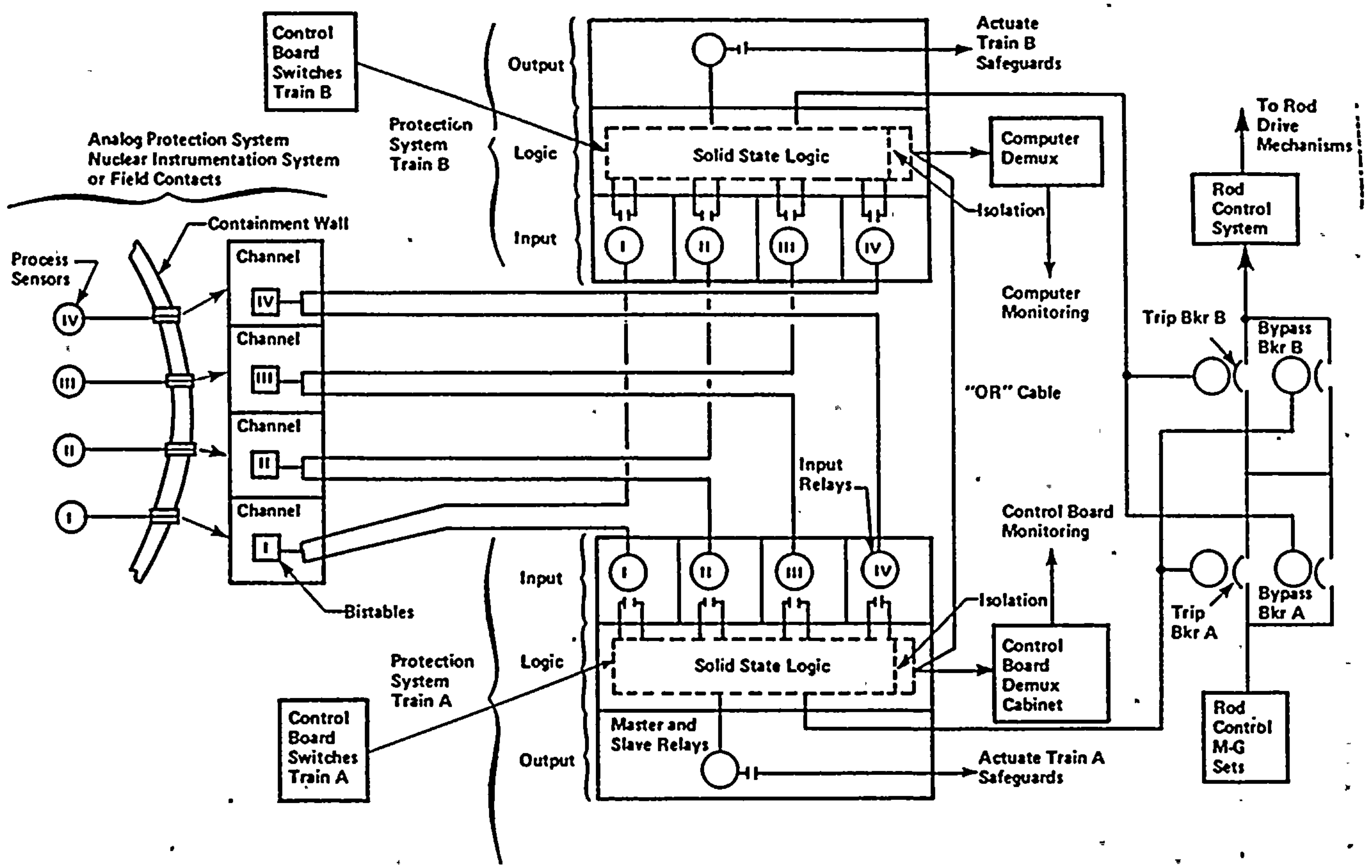
AMENDMENT NO. 15

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

125 VOLT D-C, 250 VOLT D C AND
120 VOLT A C ONE-LINE DIAGRAM

FIGURE 8.1.3-3

ATTACHMENT 2

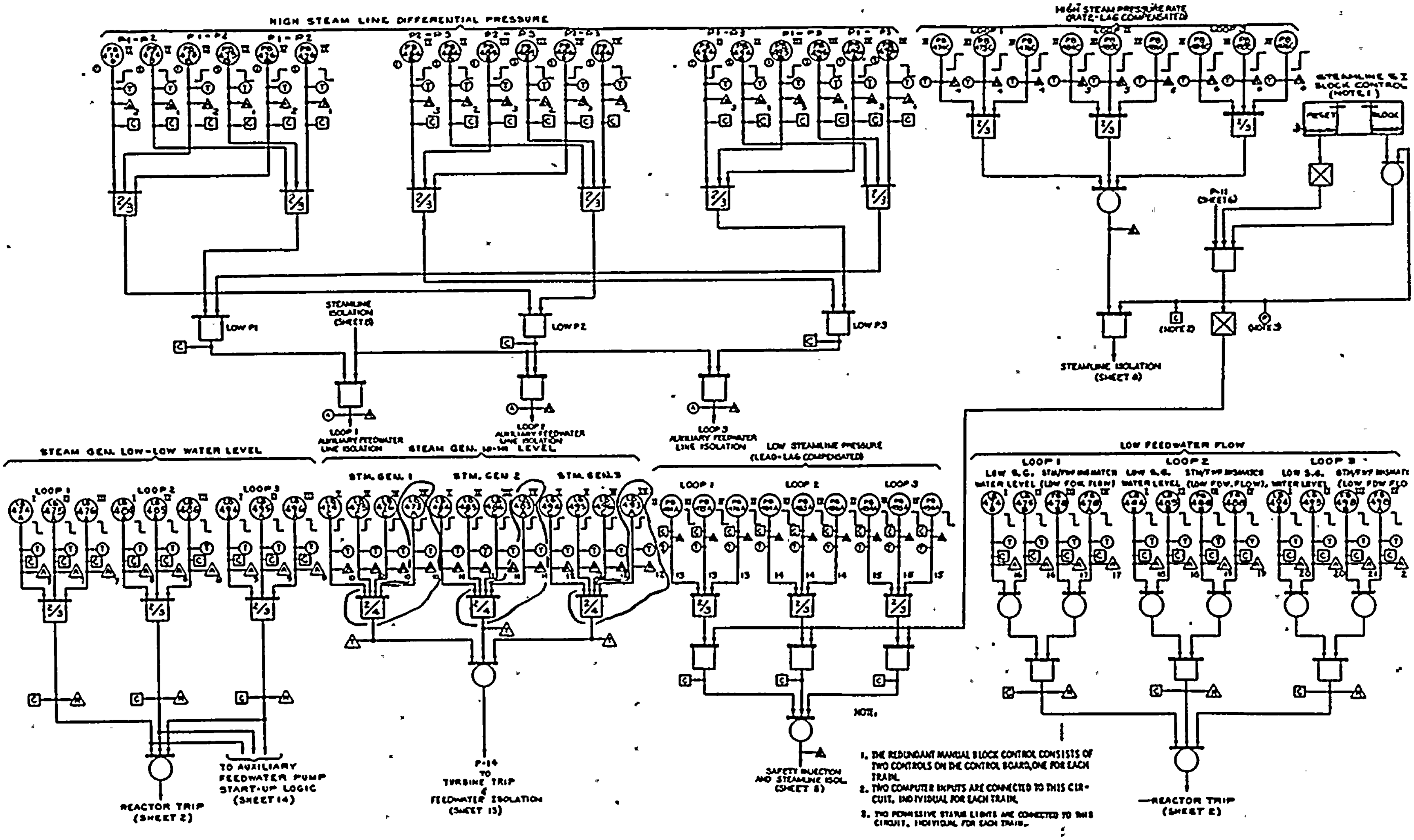


SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

PROTECTION SYSTEM BLOCK DIAGRAM

FIGURE 7.1.1-1

ATTACHMENT 3



NOTE:

1. THE REDUNDANT MANUAL BLOCK CONTROL CONSISTS OF TWO CONTROLS ON THE CONTROL BOARD, ONE FOR EACH TRAIN.
2. TWO COMPUTER INPUTS ARE CONNECTED TO THIS CIRCUIT, INDIVIDUAL FOR EACH TRAIN.
3. TWO PERMISSIVE STATUS LIGHTS ARE CONNECTED TO THIS CIRCUIT, INDIVIDUAL FOR EACH TRAIN.

AMENDMENT NO. 21

SHEARON HARRIS NUCLEAR POWER PLANT
Carolina Power & Light Company
FINAL SAFETY ANALYSIS REPORT

STEAM GENERATOR TRIP SIGNALS
FIGURE 7.2.1-1
SHEET
7 OF 15

CP&L

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USNRC-DS

1987 OCT 20 A 9:55

HARRIS NUCLEAR PROJECT
P.O. Box 165
New Hill, NC 27562

OCT 15 1987

File Number: SHF/10-13510C
Letter Number: HO-870514 (O)

U.S. Nuclear Regulatory Commission
ATTN: NRC Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT UNIT 1
DOCKET NO. 50-400
LICENSE NO. NPF-63
LICENSEE EVENT REPORT 87-054-00

Gentlemen:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence and is in accordance with the format set forth in NUREG-1022, September, 1983.

Very truly yours,

R A Watson

R. A. Watson
Vice President
Harris Nuclear Project

RAW:ddl

Enclosure

cc: Dr. J. Nelson Grace (NRC - RII)
Mr. B. Buckley (NRR)
Mr. G. Maxwell (NRC - SHNPP)

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