Attachment 5

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of LONG ISLAND LIGHTING COMPANY (Shoreham Nuclear Power Station, Unit 1)	)	Docket No. 50-322-1 (OL)

AFFIDAVIT OF ROBERT G. LaGRANGE
IN RESPONSE TO ALAB-788

I, Robert G. LaGrange, depose and say:

1. I am a Section Leader in the Equipment Qualification Branch, within the Division of Engineering, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission. A statement of my Professional Qualifications is attached. This affidavit is submitted in response to that portion of ALAB-788 dealing with "environmental qualification."

2. In ALAB-788 issued by the Atomic Safety and Licensing Appeal Board on October 31, 1984, the Appeal Board required the NRC Staff to advise the Licensing Board whether any non-safety related electrical equipment at Shoreham falls within the category defined by 10 CFR §50.49(b)(2) and, if so, the basis for the Staff's approval. (ALAB-788 at slip op. 105)

3. In compliance with the Appeal Board's requirements, the Board's attention is invited to Section 3.11.3 of the Shoreham SSER 7, issued in

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September, 1984. In particular, Section 3.11 and specifically Section 3.11.3.1 of SSER 7, which was prepared under my supervision, and with which I concurred, describes the Staff's determination that no equipment at Shoreham falls into the category defined by 10 CFR §50.49(b)(2). Additionally, the basis for the Staff's determination in this regard is fully set forth in Section 3.11.3.1. In Section 3.11.3.1 we discussed the performance of a control systems failure study, a high energy line break/control system failure analysis, and the electrical isolation design philosophy at Shoreham. The staff has reviewed these areas and has found them to be acceptable as documented in Section 7.7 of SSER 4 and Section 7.6.6 of the SER. One of the purposes of these studies was to identify nonsafety-related equipment whose failure could affect the satisfactory accomplishment of safety functions by safety-related equipment. The resolution of these issues provides a sufficient basis to conclude that there is no equipment that falls into the category defined by 10 CFR §50.49(b)(2). I hereby certify (1) that the statements contained therein are true and correct to the best of my knowledge and belief, and (2) I know of no equipment at Shoreham that falls within the category of equipment described in 10 CFR § 50.49 (b)(2).

Robert G. LaGrange
Robert G. LaGrange

Subscribed and sworn to before me
this 13th day of November, 1984

Malinda L. McDonald
Notary Public

My commission expires: 7/1/86

PROFESSIONAL QUALIFICATIONS

OF

ROBERT G. LaGRANGE

I am Section Leader of the Environmental Qualification Section of the Equipment Qualification Branch, Division of Engineering, Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission. I am responsible for planning, organizing and directing the activities of the section in performing technical reviews, analyses and evaluations of the adequacy of the environmental qualification of electrical and mechanical equipment whose failure, due to such environmental conditions as temperature, humidity, pressure and radiation, could adversely affect the performance of safety systems. I was previously a Senior Mechanical Engineer in the Seismic and Dynamic Loads Qualification Section of the Equipment Qualification Branch. My duties and responsibilities involved the review and evaluation of the structural integrity, operability and functional capability of safety related mechanical and electrical equipment under all normal, abnormal, and accident loading conditions, and in the event of seismic occurrences and other pertinent dynamic loads. Prior to my positions in the Equipment Qualification Branch, I was an Applied Mechanics Engineer in the Engineering Branch, Division of Operating Reactors. My duties and responsibilities included the review, analysis and evaluation of structural and mechanical aspects of safety issues related to reactor facilities licensed for power operation.

I have a B.S. degree in Mechanical Engineering from the University of Maryland (1972) and have done graduate work at both the University of Maryland and George Washington University.

Prior to my joining the NRC, I was associated with Bechtel Power Corporation as a Group Leader in the piping stress analysis group. My duties and responsibilities included performing and supervising stress analyses of nuclear power plant piping, and related activities, with emphasis on seismic analysis.

ATTACHMENT 6

LILCO LONG ISLAND LIGHTING
LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

SNRC-761

August 27, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SER Issue No. 47 - Control System Failures
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Dear Mr. Denton:

As stated in section 7.7 of Supplement No. 1 to the Shoreham Safety Evaluation Report (SER) the Long Island Lighting Company committed to conduct a review to demonstrate that failures or malfunctions of power sources or sensors providing power or signals to two or more control systems will not result in consequences outside the bounds of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.

In fulfillment of this commitment, enclosed are forty (40) copies of a report entitled "Control System Failures Evaluation Report". This report concludes that, although new transient category events can be postulated by considering these failures or malfunctions, the net effects have been positively determined to be less severe than those of the original, conservative Chapter 15 events.

The submittal of this report to the staff should be sufficient to completely close SER Issue Number 47.

Should you have any questions, please contact this office.

Very truly yours,

Original signed by

J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

RWG:mp

Enclosure

cc: J. Higgins
All parties

CONTROL SYSTEMS FAILURES
EVALUATION REPORT

AUGUST 1982

PREPARED

FOR

LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION

PREPARED
BY

P. R. SCHERER
GENERAL ELECTRIC COMPANY, NUCLEAR ENERGY BUSINESS OPERATIONS
San Jose, California 95125

Approved:

T. R. Wortham, Manager, Technical Licensing
Nuclear Control and Instrumentation Department

E. C. Eckert, Manager - Plant Transient Performance Engineering
Nuclear Power Systems Engineering Department

P. A. Bohm, Senior Licensing Engineer
Safety and Licensing Operation

7-0141

CONTENTS

ILLUSTRATIONS

CONTROL SYSTEMS FAILURES
EVALUATION REPORT
FOR THE SHOREHAM NUCLEAR POWER STATION

1.0 OBJECT

This document constitutes:

- An analysis in response to the NRC concern that the failures of power sources or sensors which provide power or electrical signals to multiple control systems could result in consequences outside the bounds of the Shoreham Final Safety Analysis Report (FSAR) Chapter 15 analyses and beyond the capability of operators or safety systems.

- A positive demonstration that adequate review and analysis has been performed to ensure that despite such failure the Shoreham FSAR Chapter 15 analyses are bounding, and no consequence beyond the capability of operators on safety systems would result.

A comprehensive approach was developed to analyze the control systems capable of affecting reactor water level, pressure or power in the Shoreham plant.

This report with its attachments was prepared by the General Electric Company for the Long Island Lighting Company (LILCO) with a significant technical contribution from the Stone & Webster Engineering Corporation (SWEC).

2.0 CONCLUSIONS

This report, supplemented by the existing FSAR Chapter 15 transient analyses, documents an evaluation of the Shoreham Nuclear Power Station for system interaction by electrical means. The conclusion of this evaluation is that previously reported limits of minimum critical power ratio (MCPR), peak vessel and main steamline pressures, and peak fuel cladding temperature for the expected operational occurrence category of events would not be exceeded as a result of common power source or sensor failures. Although new transient category events have been postulated as a result of this study, the net effects have been positively determined to be less severe than those of the original, conservative, Chapter 15 events. It should be noted that this study uses the event - consequence logic of the Chapter 15 analysis, but starts the logic chain from a specific source (e.g., a single bus failure) rather than a system condition (e.g., feedwater runout). By approaching the study in this manner, a great deal of confidence can be placed in the study conclusions. The approach validated itself by uncovering previously unanalyzed interactions. The soundness of the total plant design is demonstrated by its being tolerant of these interactions.

3.0 ANALYSIS METHODOLOGY

The electrical control systems failure analysis was conducted in the following manner by GE and the SWEC:

ACTIVITY	ASSIGNED TO
• DEFINE BUS STRUCTURE	SWEC
• DEFINE CONTROL SYSTEMS	SWEC & GE
• IDENTIFY LOADS	SWEC & GE
• DETERMINE CRITICAL LOADS	SWEC & GE
• SUMMARIZE CRITICAL LOADS	GE
• ANALYZE COMBINED EFFECTS	GE
• COMPARE RESULTS TO CHAPTER 15	GE
• ANALYZE EXCEPTIONS	GE
• MODIFY/AUGMENT CHAPTER 15 IF NECESSARY	GE

3.1 DEFINE BUS STRUCTURE

This step established the potential sources for system interaction by electrical means. Bus trees (see Figures 1 and 2) were constructed using one-line diagram information to show power distribution from the highest level not previously analyzed (the highest level previously analyzed is the loss of offsite power) down to the lowest level of plant distribution (Motor Control Center's, instrument busses, etc.).

3.2 DEFINE CONTROL SYSTEMS

This step established the scope of control systems to be analyzed. A complete list of Shoreham plant systems and subsystems was compiled. This list was then reviewed to confine the analysis to only those systems with the potential to affect reactor pressure, water level, or power.

To ensure that all necessary systems were considered, certain elimination criteria were established that documented the justification for not analysing that system further. If there was any uncertainty as to whether or not a system met the criteria, it was retained for further analysis. Those systems that met the criteria for elimination were removed from the complete system list to produce the final list of control systems for analysis. This final list, reviewed by GE and SWEC, is shown as follows:

3.2 DEFINE CONTROL SYSTEMS (Continued)

MPL	SYSTEMS	MPL	SYSTEMS
B21	Nuclear Boiler System	N42	Hydrogen Seal System
B31	Reactor Recirculation	N43	Generator Cooling
C11	CRD Hydraulic Control System	N44	Air Removal
C32	Feedwater Turbine	N45	Generator Hydrogen & CO_2 Purge
C51	Neutron Monitoring	N51	Main Generator Excitation
D11	Process Radiation Monitor System	N62	Off Gas
D21	Area Radiation Monitor System	N71	Circulating Water
G33	Reactor Water Cleanup	P41	Service Water
N11	Main Steam	P42	RB Closed Cooling Water System
N21	Condensate	P43	TB Closed Cooling Water System
N32	Turbine Control	P50	Compressed Air
N34	Lube Oil	P71	Low Conductivity Drains
N35	Moisture Extraction	Z93	Primary Containment Instrumentation

3.3 IDENTIFY LOADS

This step provided the data base necessary to determine which electrical loads were to be analyzed. A set of load tables comprised of all electrical loads of the control systems in Paragraph 3.2 was assembled by GE and SWEC, each providing information on the loads within their respective scope of supply.

Each load was listed with its power bus source, its unique Master Parts List number, circuit description, and failure mode on power loss with primary and secondary effects. A sample of a load table is included in Appendix C.

3.4 DETERMINE CRITICAL LOADS

This step constituted the first analytical step in sorting out the loads with the potential for initiating events affecting reactor pressure, water level and power. The elimination criteria established earlier for the system list was refined in Appendix B for use in the component review for determining which individual loads were worthy of further consideration or could be deleted from the analysis. If there was any uncertainty as to whether or not a load met the elimination criteria it was retained for further analysis. The code associated with an elimination criterion was assigned to each eliminated load in the load tables discussed in the previous step.

3.5 SUMMARIZE CRITICAL LOADS

The non-critical loads were deleted from the load tables, and the remaining loads are grouped together by their common power busses. These tables are shown in Appendix A.

3.6 ANALYZE COMBINED EFFECTS

This step provided the basis for determining the worst case combinations of load and system failures that are credible events considering the interconnection by power distribution. Using the combined effects at the lowest level bus as a starting point, the next higher bus was postulated to fail and the total effects at that level analyzed. This process was continued up to the highest bus level. The combined effects at the lowest bus level are included in the Appendix A tables. Worst case effects at the higher levels are summarized in Section 4. The combined effects at intermediate bus levels less severe than their associated higher bus combined effects were analyzed but not included in Section 4. The intermediate level combined effect analysis is already represented in the higher bus analysis.

3.7 COMPARE RESULTS TO CHAPTER 15

This step evaluated the consequences of all potential system interaction events initiated by electrical means. A review of the information in the Appendix A tables was conducted in the course of developing the bus summaries of Section 4. At each bus level of the combined effects analysis, the review evaluated the effects as being bounded by a specific Chapter 15 transient analysis or not. Section 4 includes these evaluations considering the worst case effects.

3.8 ANALYZE EXCEPTIONS

The purpose of this step was to determine if a failure scenario not directly covered by a Chapter 15 transient analysis would be bounded by one with more severe effects. The cases of this type are included in the Section 4 descriptions of worst case failures.

3.9 MODIFY/AUGMENT CHAPTER 15 IF NECESSARY

This step was not necessary in the Shoreham analysis.

4.0 BUS LOSS SUMMARY RESULTS AND CHAPTER 15 COMPARISONS

AC Bus

1A (4.16KV) Loss of this bus causes the loss of power to condensate booster pump A and reactor recirculation pump A. There is also a potential main turbine trip due to the circulating water pump A loss and its subsequent effect on condenser vacuum. Since a reduction of reactor recirculation flow would immediately start reducing reactor power, an immediate or delayed turbine trip would produce an equal or less severe transient than the turbine trip event of Chapter 15. Therefore this event is bounded.

1B (4.16KV) The effects of the loss of this bus are similar to those of the loss of Bus 1A.

11 (4.16KV) Loss of this bus will cause condensate pump A and circulating water pump C to be inoperative. The loss of the condensate pump will initiate reactor recirculation flow to run back and reduce reactor power corresponding to 67 percent of rated feedwater flow. In addition, a loss of feedwater heating of less than 10°F will occur, but this effect will be nullified by the recirculation runback. In the event that circulating water pump A or B is in the backwash operation, the loss of circulating water pump C may cause pump D to flow back and effectively reduce the circulating water flow to a one-pump operation; and the condenser back pressure may rise rapidly leading to a main turbine trip. The ensuing pressure excursion may even reach the bypass closure trip setpoint. However, this event will take place at reduced reactor power and it is bounded by the turbine trip without bypass transient already analyzed in FSAR Chapter 15.

Loss of the associated lower busses fed by Bus 11 will produce some or all of the following effects: Decrease in condenser vacuum, delayed main turbine trip, reduction in feedwater flow, and reduction in reactor recirculation flow.

The worst case reduction in feedwater temperature has been determined to be no more than 10°F. This reduction in feedwater heating will increase reactor power by less than three percent nuclear boiler rated (NBR) power.

The worst case scenario is the unlikely event of a loss of feedwater heating and a delayed turbine trip. A computer analysis was performed to determine the reactor parameters as a consequence of a turbine trip at approximately 103 percent of initial power. The results yielded a minimum critical power ratio (MCPR) of 1.10 and a maximum dome pressure of 1197 psia which is less severe than the most limiting transient analyzed in FSAR Chapter 15. This event is then, although previously not analyzed for the Shoreham plant, still bounded by existing analyses.

4.0 BUS LOSS SUMMARY RESULTS AND CHAPTER 15 COMPARISONS (Continued)

AC Bus

12 (4.16KV)	The effects of the loss of this bus are similar to those of the loss of Bus 11.
101/102 (4.16KV) Emergency	The loss of either of these busses will cause a single channel trip from the APRM circuitry to the reactor protection system which produces no transient.
103 (4.16KV) Emergency	Loss of this bus will cause a decrease in reactor recirculation flow and a lock of the feedwater pumps at-last-speed setting. An increase in level would ensue terminated by the level 8 feedwater pump and main turbine trip. This event is similar to and bounded by the feedwater runout event analyzed in Chapter 15.

DC Bus

1R42-BA N1 & N2	The worst case effect of the loss of either of these battery buses is a main turbine trip with no additional complications which is bounded by Chapter 15 load rejection analysis.

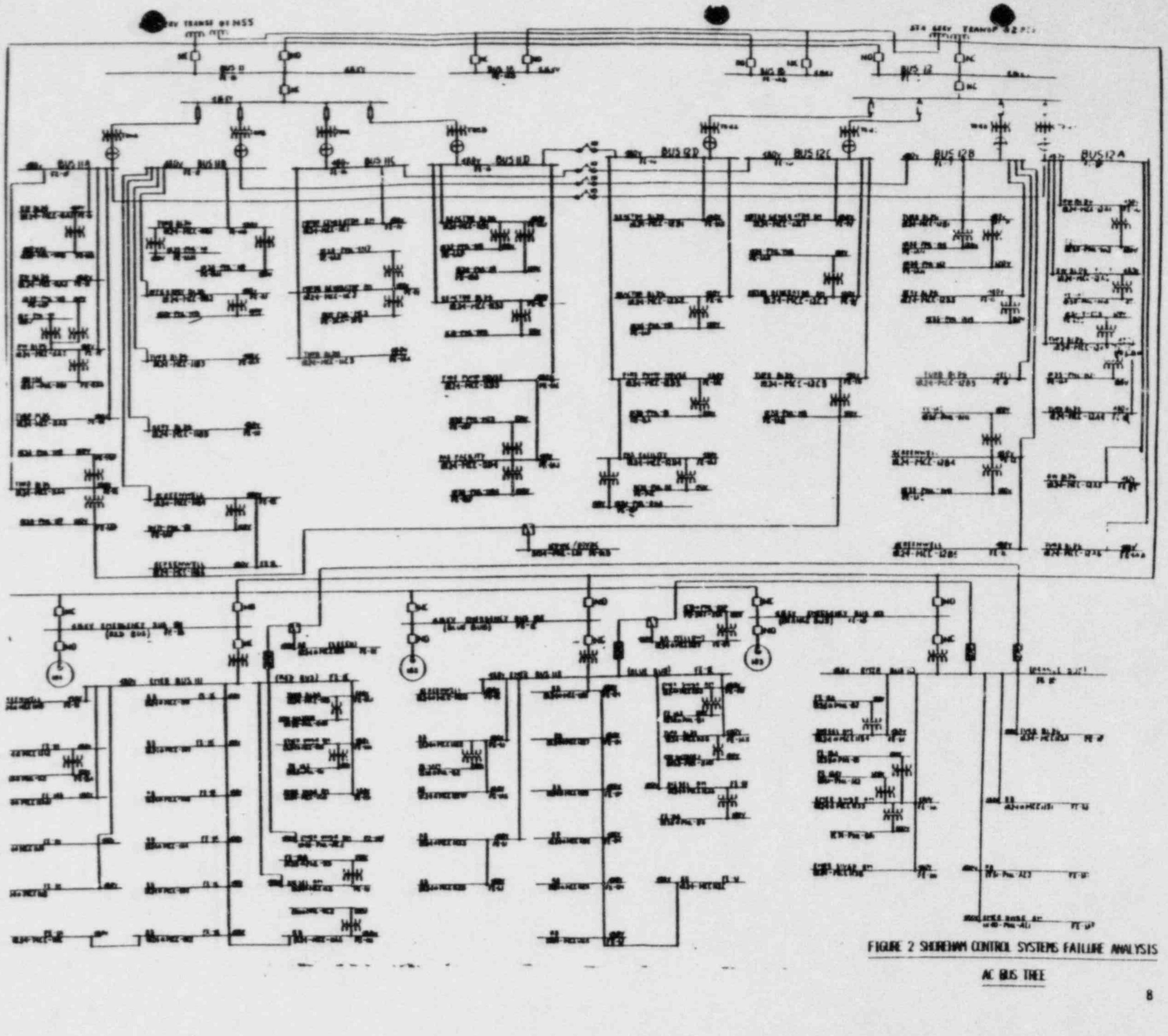

FIGURE 2 SHOREHAM CONTROL SYSTEMS FAILURE ANALYSIS

AC BUS TREE

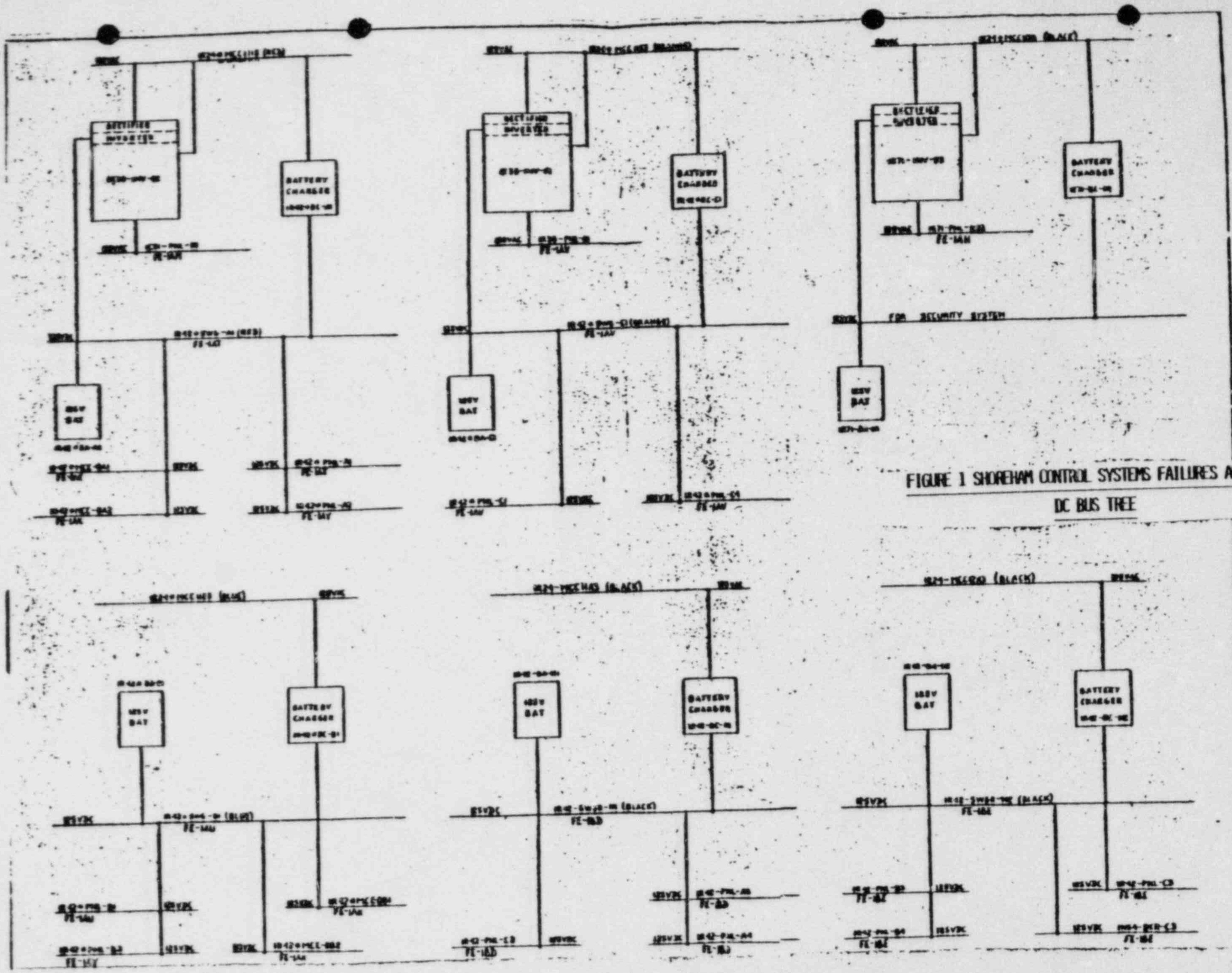

FIGURE 1 SHOREHAM CONTROL SYSTEMS FAILURES ANALYSIS

DC BUS TREE

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
AC BUS 1A	RECIRC	GENERATOR DRIVE MOTOR – S001A	LOSE GENERATOR DRIVE MOTOR – S001A	RUN BACK TO 66% POWER	REDUCTION OF FEEDWATER FLOW TO 67% OF RATED
	CIRC WTR	CIRCULATIING WATER PUMP A (PUMP B · BUS 1B) (PUMP C · BUS 11) (PUMP D · BUS 12)	DECREASE CONDENSER VACUUM	SLIGHT DECREASE IN CONDENSER VACUUM	RECIRCULATION RUNBACK TO 66% REACTOR POWER SLIGHT DECREASE IN CONDENSER VACUUM
	CONDENSATE	CONDENSATE BOOSTER PUMP A (PUMP B · BUS 1B)	REDUCTION OF FEEDWATER TO 67% OF RATED	RUNBACK TO 66% REACTOR POWER	

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
AC BUS 1B	RECIRC	GENERATOR DRIVE MOTOR – S001B	LOSE GENERATOR DRIVE MOTOR – S001B	RUN BACK TO 66% POWER	REDUCTION OF FEEDWATER FLOW TO 67% OF RATED
	CIRC. WTR	CIRCULATING WATER PUMP B (PUMP A BUS 1A) (PUMP C BUS 11) (PUMP D BUS 12)	DECREASE CONDENSER VACUUM	SLIGHT DECREASE IN CONDENSER VACUUM	RECIRCULATION RUNBACK TO 66% REACTOR POWER SLIGHT DECREASE IN CONDENSER VACUUM
	CONDENSATE	CONDENSATE BOOSTER PUMP B (PUMP A BUS 1A)	REDUCTION OF FEEDWATER FLOW TO 67% OF RATED	RUNBACK TO 66% REACTOR POWER	

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC
BUS 11

	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
BUS 11	CONDENSATE	CONDENSATE PUMP A (PUMP B BUS 12)	FEEDWATER REDUCEC TO 67% OF RATED	REACTOR PRESSURE VESSEL WATER LEVEL LOWER AND A 65% POWER	(SEE SECTION 4)
BUS 11	COMP. AIR	AIR COMPRESSOR A (COMPRESSOR B & C BUS 12)	COMPRESSOR A INOPERATIVE	NONE – BACKED UP BY COMPRESSORS B & C	
BUS 11	CIRC. WTR	CIRCULATION WATER PUMP C (PUMP A BUS 1A) (PUMP B – BUS 1B) (PUMP D BUS 12)	PUMP INOPERATIVE	DECREASE CONDENSER VACUUM MAIN TURBINE TRIP.	
BUS 11	SERVICE WATER TURBINE BUILDING	SERVICE WATER PUMP A (PUMPS B & C – BUS 12)	PUMP INOPERATIVE	NONE – BACKED UP BY PUMPS B & C	
BUS 11A – 1R24 MCC11A1	COMP. AIR	WASTE NEUTRAL TANK INLET AIR MOV 81	IF OPEN, WILL LOSE INSTRUMENT AIR (COMPRESSORS B & C BUS 12 AND 1R36 PNL N8)	NONE	NONE
BUS 11A – 1R24 MCC11A2	OFFGAS	DRYER TRAIN A (TRAIN B – MCC12A2)	LOSS OF DRYER TRAIN A	NONE – BACKED UP BY DRYER TRAIN B	NONE
BUS 11A – 1R24 MCC11A3	CONDENSATE	HEATER TRAIN BYPASS MOV 30	MOV FAILS AS IS	NONE	WORST CASE – DECREASE IN CONDENSER VACUUM
BUS 11A – 1R24 MCC11A3	GEN. COOLING	STATOR COOLING WATER PUMP A (PUMP B – BUS 12)	IF PUMP B NOT AVAILABLE, MAIN TURBINE TRIP	NONE – BACKED UP BY PUMP B	
BUS 11A – 1R24 MCC11A3	AIR REMOVAL	AIR EJECTOR ISOLATION MOV 45A AIR EJECTOR ISOLATION MOV 46A	MOV'S FAIL AS IS	IF IN BACKWASH, SLIGHT DECREASE IN CONDENSER VACUUM	
BUS 11A – 1R24 MCC11A3	CIRC. WATER	CONDENSER INLET MOV 32A CONDENSER DISCHARGE MOV 33A CONDENSER BACKWASH VALVE MOV 34A	IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS	DECREASE CONDENSER VACUUM DECREASE CONDENSER VACUUM	

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC BUS 11

	SYSTEM	COMPONENT DESCRIPTION		PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
BUS 11A 1R24 MCC11A4	COMP. AIR	COMPRESSOR START UP AUXILIARY OIL PUMP A		AUX OIL PUMP INOPERATIVE	UNABLE TO START AIR COMPRESSOR A	DECREASE CONDENSER VACUUM AND FEEDWATER TEMPERATURE
	OFFGAS	HOT GAS BYPASS	SOV 49A	SOV'S DE ENERGIZED	NONE BACKED UP BY TRAIN B (TRAIN B – BUS 12A)	
		LIQUID FREON	SOV 31A			
1R36 PNL N7	CONDENSATE	REACTOR FEEDWATER PUMP A DISCHARGE	SOV 42A	SOV DE ENERGIZED	SLIGHT INCREASE IN REACTOR FEEDWATER PUMP TURBINE SPEED	DECREASE CONDENSER VACUUM AND FEEDWATER TEMPERATURE
	MOISTURE EXTRACTION	GLAND STEAM EVAPORATOR DRAIN	SOV 10L	SOV–10L VALVE CLOSES	BYPASS HEATER STEAM TO CONDENSER	
		STEAM SEAL EVAPORATOR DRAIN	SOV 10H	SOV 10H VALVE OPENS	DECREASE IN FEEDWATER TEMPERATURE AND CONDENSER VACUUM	
		RADWASTE STEAM GENERATOR DRAIN	SOV 11L	SOV 11L VALVE CLOSES		
		RADWASTE STEAM GENERATOR DRAIN	SOV 11H	SOV 11H VALVE OPENS		
		1ST POINT HEATER	SOV 01AH	SOV 01 AH VALVE OPENS	DECREASE FEEDWATER TEMPERATURE AND CONDENSER VACUUM	
		1ST POINT HEATER	SOV 01AN	SOV 01 AN VALVE CLOSES		
		2ND POINT HEATER	SOV 02AH	SOV 02 AH VALVE OPENS		
		2ND POINT HEATER	SOV 02AN	SOV 02 AN VALVE CLOSES		
		3RD POINT HEATER	SOV 03AH	SOV 03 AH VALVE OPENS		
		3RD POINT HEATER	SOV 03 AN	SOV 03 AN VALVE CLOSES		
		4TH POINT HEATER	SOV 04 AH	SOV 04 AH VALVE OPENS		
		4TH POINT HEATER	SOV 04 AN	SOV 04 AN VALVE CLOSES		
		6TH POINT HEATER	SOV 06 AH	SOV 06 AH VALVE OPENS		
	COMP. AIR	AIR COMPRESSOR CONTROL CIRCUIT A (CIRCUITS B & C– 1R35 PNL N8)		LOSS OF INSTRUMENT AIR COMPRESSOR A	NONE – BACKED UP BY AIR COMPRESSORS B & C	

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC
BUS 11B

	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
1R24 MCC11B1	CIRC. WATER	CIRCULATION WATER CONDEN- INLET MOV-32C CIRCULATION WATER CONDEN- DISCHARGE MOV-33D CONDENSER BACKWASH VALVE MOV-34C	IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS	DECREASE CONDENSER VACUUM DECREASE CONDENSER VACUUM DECREASE CONDENSER VACUUM	WORST CASE – DECREASE CONDENSER VACUUM RFPT A TRIP. FEEDWATER FLOW REDUCTION TO 67% OF RATED. MINIMUM SPEED ON RECIRC A & B PUMPS – 60% REACTOR POWER
	MAIN TURBINE CONTROL	MAIN TURBINE EHC FLUID PUMP A (PUMP B – MCC12B1)	PUMP A INOPERATIVE	NONE – BACKED UP BY PUMP B	
	FEEDWATER	CONTROL SIGNAL FAIL INITIATING CONTACT	LOSS OF REACTOR FEEDWATER PUMP'S CONTROL SIGNAL WILL NOT SET AT LAST SPEED	SCRAM IF REACTOR FEED WATER PUMP'S CONTROL SIGNAL LOST	MINIMUM SPEED ON RECIRC A & B PUMPS – 50% POWER
1R36 PNL N1	RECIRC	RECIRCULATION DIVISION 1 SPEED CONTROL	RECIRCULATION PUMPS A & B MINIMUM SPEED IF IN AUTO MODE	REACTOR AT 60% POWER	
1R36 PNL N3	CONDENSATE	MINIMUM FLOW BYPASS SOV-28A	SOV DE-ENERGIZED FCV FAILS OPEN	RFP TURBINE A TRIP. FEED WATER FLOW REDUCTION TO 67% OF RATED. RECIRC RUNBACK TO 66% POWER	RFP TURBINE A TRIP. FEED WATER FLOW REDUCTION TO 67% OF RATED. RECIRC RUNBACK TO 66% POWER
1R24 MCC11B4	CIRC. WATER	CIRCULATION WATER PUMP A DISCHARGE MOV-31A CIRCULATION WATER PUMP C DISCHARGE MOV-31C	MOV'S FAIL AS IS MOV'S FAIL AS IS	FAILED CLOSED – UNABLE TO START PUMP(S) FAILED OPEN – NO EFFECT ON PUMP(S)	LOSS OF TURBINE BUILDING SERVICE WATER STRAINER BACKWASH CAPABILITY IF CIRC WATER DISCHARGE VALVES FAIL OPEN AND PUMPS STOP, UNABLE TO PREVENT BACK FLOW, DECREASE CONDENSER VACUUM – MAIN TURBINE TRIP
1R24 MCC11B8	SERVICE WATER	TURBINE BUILDING SERVICE WATER INLET MOV-113A	MOV FAILS AS IS – NORMALLY CLOSED. LOSS OF STRAINER BACKWASH CAPABILITY	WORST CASE – MAIN TURBINE AND RFP TURBINE TRIP AFTER MANY HOURS	WORST CASE – MAIN TURBINE AND RFP TURBINE TRIP AFTER MANY HOURS

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC BUS 12

BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
AC BUS 12	CONDENSATE	CONDENSATE PUMP B (PUMP A – BUS 11)	FEEDWATER REDUCED TO 67% OF RATED	REACTOR PRESSURE VESSEL WATER LEVEL LOWER AND AT 66% POWER	(SEE SECTION 4)
	COMP. AIR	INSTRUMENT AIR COMPRESSOR B INSTRUMENT AIR COMPRESSOR C (COMPRESSOR A ON BUS 11)	AIR COMPRESSORS B & C INOPERATIVE	NONE – BACKED UP BY COMPRESSOR A	
	CIRC. WATER	CIRCULATION WATER PUMP D	PUMP INOPERATIVE	DECREASE CONDENSER VACUUM. MAIN TURBINE TRIP	
	SERVICE WATER TURBINE BUILDING	SERVICE WATER PUMP B SERVICE WATER PUMP C	PUMPS INOPERATIVE	REDUCED TURBINE COOLING WATER. MAIN TURBINE TRIP AFTER SOME TIME	
BUS 12A 1R24 MCC12A2	OFFGAS	DRYER TRAIN B (TRAIN A–MCC11A2)	LOSS OF DRYER TRAIN B	NONE – BACKED UP BY TRAIN A	NONE
1R24 MCC12A3	GEN. COOLING	STATOR COOLING WATER PUMP B (PUMP A – BUS 11)	PUMP INOPERATIVE IF PUMP A NOT AVAILABLE MAIN TURBINE TRIP	NONE – BACKED UP BY PUMP A	WORST CASE – DECREASE CONDENSER VACUUM
	CIRC. WATER	CIRCULATION WATER CONDENSATE INLET MOV 32B CIRCULATION WATER CONDENSATE DISCHARGE MOV 33B CONDENSER BACKWASH VALVE MOV 34B	IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS	DECREASE IN CONDENSER VACUUM DECREASE IN CONDENSER VACUUM DECREASE IN CONDENSER VACUUM	
1R24 MCC12A4	COMP. AIR	COMPRESSOR AUXILIARY OIL PUMP B COMPRESSOR AUXILIARY OIL PUMP C	PUMPS INOPERATIVE	COMPRESSORS B & C WILL NOT START IF DEMAND ON AIR SYSTEM REQUIRES	WORST CASE – DECREASE CONDENSER VACUUM
	AIR REMOVAL	AIR EJECTOR ISOLATION MOV 45B AIR EJECTOR ISOLATION MOV 46B	MOV'S FAIL AS IS	IF IN BACKWASH, SLIGHT DECREASE IN CONDENSER	
	OFFGAS	HOT GAS BYPASS SOV 49B LIQUID FREON SOV 31B	SOV'S DE ENERGIZE	NONE – BACKED UP BY TRAIN A (TRAIN A–BUS 11A)	

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC
BUS 12

Bus	System	Component Description	Primary Effect	Secondary Effect	Combined Effects
BUS 12B / 1R24 MCC12B1	MAIN TURBINE CONTROL	MAIN TURBINE EHC FLUID PUMP B (PUMP A – MCC11B1)	PUMP INOPERATIVE	NONE – BACKED UP BY PUMP A	WORST CASE – DECREASE IN CONDENSER VACUUM RFP TURBINE TRIP. FEED WATER FLOW REDUCTION TO 67% OF RATED, MINIMUM SPEED ON RECIRC A & B PUMPS – 60% REACTOR POWER
	CIRC. WATER	CONDENSER BACKWASH VALVE MOV-34D	IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS	DECREASE CONDENSER VACUUM	
		CIRCULATION WATER CONDENSATE INLET MOV-32D	IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS	DECREASE CONDENSER VACUUM	
		CIRCULATION WATER CONDENSATE DISCHARGE MOV-33C	IF IN BACKWASH, REDUCE FLOW TO 2 QUADRANTS	DECREASE CONDENSER VACUUM	
1R36 PNL N2	RECIRC	RECIRCULATION DIVISION II SPEED CONTROL	RECIRCULATION A & B MINIMUM SPEED IF IN AUTO MODE	RUN BACK TO 60% POWER	RECIRCULATION PUMPS A & B AT MINIMUM SPEED–60% REACTOR POWER
1R36 PNL N4	CONDENSATE	MINIMUM FLOW BYPASS SOV-28B	SOV DE-ENERGIZED FCV FAILS OPEN	RFP TURBINE TRIP. FEED WATER FLOW REDUCTION TO 67% OF RATED. RECIRC RUNBACK TO 65% REACTOR POWER	RFP TURBINE TRIP. FEED WATER FLOW REDUCTION TO 67% OF RATED. RECIRC RUNBACK TO 65% REACTOR POWER
1R24 MCC12B4	CIRC. WATER	CIRCULATION WATER PUMP DISCHARGE MOV-31B CIRCULATION WATER PUMP DISCHARGE MOV-31D	MOV'S FAIL AS IS MOV'S FAIL AS IS	FAILED CLOSED – UNABLE TO START PUMP(S). FAILED OPEN – NO EFFECT ON PUMP(S)	LOSS OF TURBINE BUILDING SERVICE WATER STRAINER BACKWASH CAPABILITY IF CIRC WATER DISCHARGE VALVES FAIL OPEN AND PUMPS STOP, UNABLE TO PREVENT BACK FLOW, DECREASE CONDENSER VACUUM – MAIN TURBINE TRIP
1R36 PNL N10	CIRC. WATER	STRAINER – S 81A STRAINER – S 81B	STRAINER MOTOR INOPERATIVE	UNABLE TO BACKWASH – MAIN TURBINE TRIP AFTER SEVERAL HOURS	MAIN TURBINE TRIP
1R35 PNL N11	CIRC. WATER	CIRCULATION WATER PUMPS CONTROL CIRCUIT INTERLOCK	CIRCULATION WATER PUMPS A, B, C & D CANNOT RE-START IF ANY SHOULD STOP	NONE	NONE

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC BUS 12

AC BUS 12	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
1R24 MCC1284					
1R24 MCC1286	SERVICE WATER	TURBINE SERVICE WATER STRAINER INLET MOV-113B	MOV FAILS AS IS – NORMALLY CLOSED. LOSS OF STRAINER BACKWASH CAPABILITY	WORST CASE – MAIN TURBINE AND RFP TURBINE TRIP AFTER MANY HOURS	WORST CASE – MAIN TURBINE AND RFP TURBINE TRIP AFTER MANY HOURS

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC
BUS 12

	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
BUS 12C 1R24 MCC12C3 1R35 PNL N8	COMP. AIR	AIR COMPRESSOR CONTROL CIRCUIT PUMP B AIR COMPRESSOR CONTROL CIRCUIT PUMP C (CIRCUIT A – 1R36 PNL-N7)	LOSE INSTRUMENT COMPRESSORS B & C	NONE – BACKED UP BY COMPRESSOR A	DECREASE CONDENSER VACUUM AND FEEDWATER TEMPERATURE
	STEAM SYSTEM	STEAM SUPPLY REACTOR FEEDWATER PUMP TURBINE SOV-30A STEAM SUPPLY REACTOR FEEDWATER PUMP TURBINE SOV-30B	VALVES INOPERABLE	NONE – STEAM FLOW TO RFPT'S AT NORMAL OPERATION UNAFFECTED	
	CONDENSATE	OPERATES FEEDWATER DISCHARGE VALVE NRV-42B SOV-42B	SOV DE-ENERGIZED	SLIGHT INCREASE IN REACTOR FEEDWATER PUMP TURBINE SPEED	
	MOISTURE EXTRACTION	1ST STAGE DRAIN TANK DRAINS TO CONDENSER SOV-7AH SOV-7AL SOV-7BH SOV-7BL	DUMPS STEAM TO CONDENSER	DECREASE CONDENSER VACUUM	
		2ND STAGE DRAIN TANK DRAINS TO CONDENSER SOV-8AH SOV-8AL SOV-8BH SOV-8BL		DECREASE CONDENSER VACUUM	
		MOISTURE SEPARATOR & REHEATERS DRAIN TANK DRAINS TO CONDENSER SOV-9AH SOV-9AL SOV-9BH SOV-9BL			
		1ST POINT HEATER DRAIN SOV-1BH 1ST POINT HEATER DRAIN SOV-1BN 2ND POINT HEATER DRAIN SOV-2BH 2ND POINT HEATER DRAIN SOV-2BN 3RD POINT HEATER DRAIN SOV-3BH 3RD POINT HEATER DRAIN SOV-3BN	BYPASS HEATER STEAM TO CONDENSER	DECREASE FEEDWATER TEMPERATURE AND CONDENSER VACUUM	

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC BUS 12	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
BUS 12C 1R24 MCC12C3 1R35 PNL N8	MOISTURE EXTRACTION	4TH POINT HEATER DRAIN SOV-4BH 4TH POINT HEATER DRAIN SOV-4BN 5TH POINT HEATER DRAIN SOV-5BH	BYPASS HEATER STEAM TO CONDENSER	DECREASE FEEDWATER TEMPERATURE AND CONDENSER VACUUM	

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

AC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
AC BUS 101 – BUS 111 – 1R24 MCC1116 (BACK UP TO MGS B)	NEUTRON MONITORING	AVERAGE POWER RANGE MONITOR CHANNELS A, C, E ROD BLOCK MONITOR CHANNEL A	HALF SCRAM TRIP	IF ALTERNATE CHANNELS ALSO TRIP, REACTOR SCRAM	½ AVERAGE POWER RANGE MONITOR SCRAM – DIV. I
AC BUS 102 – BUS 112 – 1R24 MCC1126 (BACK UP TO MGS A)	NEUTRON MONITORING	AVERAGE POWER RANGE MONITOR CHANNELS B, D, F ROD BLOCK MONITOR CHANNEL B	HALF SCRAM TRIP	IF ALTERNATE CHANNELS ALSO TRIP, REACTOR SCRAM	½ AVERAGE POWER RANGE MONITOR SCRAM – DIV. II
AC BUS 103 – BUS 113 – 1R24 MCC1133 – 1R36 INV 01 125 VDC FROM INVERTER – 1R36 PNL 01 (NO BATTERY BACK UP)	FEEDWATER	REACTOR FEEDWATER PUMPS CONTROL SIGNAL CIRCUITRY	FEEDWATER PUMPS REMAIN AT LAST SPEED UNLESS 1R36-PNL-N1 IS LOST – THEN REACTOR FEEDWATER PUMP'S RUN DOWN	LOAD FOLLOWING MIS MATCH WILL CAUSE MAIN TURBINE TRIP	RECIRCULATION PUMPS A & B AT MINIMUM SPEED – 60% POWER
	REACTOR RECIRC	RECIRCULATION CONTROL SIGNAL CIRCUITRY	IF IN AUTO MODE, PUMPS WILL RUN BACK TO MINIMUM SPEED (APPROXIMATELY 60% POWER)	(SEE REACTOR FEEDWATER CONTROL CIRCUIT ABOVE)	

APPENDIX A – BUS TABLES

SHOREHAM CONTROL SYSTEM FAILURE ANALYSIS

	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
DC BUS 1R42 BA N1 125 VDC 1R42 PNL A4	RFPT CONTROL	REACTOR FEEDWATER PUMP TURBINE A PANEL 2 EHC (REACTOR FEEDWATER PUMP TURBINE B – PANEL 1R42-PNL-B4)	CONTINUES IF AT SPEED	NONE, IF AT SPEED	MAIN TURBINE TRIP
1R42 PNL C2	MAIN TURBINE CONTROL	MAIN TURBINE PANEL 1 EHC	MAIN TURBINE TRIP	MAIN TURBINE TRIP	
DC BUS 1R42 BA N2 125 VDC 1R42 PNL B4	FEEDWATER	HIGH LEVEL B TRIP CIRCUIT	2 OF 3 HIGH LEVEL TRIP INTACT, BUT B TRIPPED		NONE
	RFPT CONTROL	REACTOR FEEDWATER PUMP TURBINE B PANEL 3 EHC (REACTOR FEEDWATER PUMP TURBINE A – PANEL 1R42-PNL-A4)	CONTINUES IF NOT AT SPEED	NONE, IF AT SPEED	

APPENDIX B

ELIMINATION CRITERIA

Elimination Criterion *	Basis
N1	Components whose failure effects are clearly bounded by a dominant failure effect on the same bus can be eliminated by inspection. An example would be the loss of several trips such as feedwater turbine overspeed trip on the same bus as the solenoid that controls all remote trips. The solenoid loss is clearly the dominant effect. Also in the case of identical components, only one of the components on that bus need be listed.
N2	Instrumentation with no direct or indirect controlling function or passive input (such as a permissive) into control logic. Instrumentation and other dedicated inputs to the process computer, as well as the computer itself, may be excluded. Operator actions as a result of indications are not considered control functions for the control systems failure analysis.
N3	Control systems and controlled components (pumps, valves) which have no direct or indirect interaction with reactor operation/parameters. Examples are communications, most unit heaters and controls, lighting controls, ventilation control systems for exterior buildings, machine shop equipment, refueling or maintenance equipment controls, etc.
N4	Control systems and controlled components (pumps, valves) that do interact or interface with reactor operating systems but which cannot affect the reactor parameters (water level, pressure or reactivity) either directly or indirectly. Examples are: some offgas components, area radiation monitors.
N5	Systems which are not used during normal power operation. For example, eliminate start-up, shutdown or refueling systems not used during normal operation.
N6	Some lube oil pumps are powered from AC busses but have a back-up pump powered from a DC source. Since a single electrical failure cannot disable the lube oil function these components can be eliminated from the analysis.
Y	Requires further analysis.

* In some cases more than one of these criteria may apply.

APPENDIX C

SAMPLE LOAD TABLE

Power Bus (AE Designation)	System (Including MPL #)	System Control and Instrumentation Loads on Bus	GENERAL EFFECTS: Limitations on the system's capability to perform its principal function due to loss of bus	SPECIFIC EFFECTS OF BUS TO SYSTEM: Effect on System Subfunctions	Effect on other Systems	CODE	Effects Related to ... on REACTOR WATER LEVEL, PRESSURE or CPR	Inputs Required from other systems and effect of LOSS of Input
*1N34-PNL-L04	N34	1N34-LS002A	Pump 1N34-P-103 will be operable from 1N34-LS003	None	None	B	None	
		1N34-LS002B		"	"	"	"	
*1N34-PNL-L04 has two power supplies from 1R33-PNL-4D (feed from Bus 11) and 1R33-PNL-4D (feed from Bus 1L) with an automatic switch. Therefore, panel will not lose power if one bus is interrupted (FE-3C0, PE-3L1)		1N34-LT002A		"	"	"	"	
		1N34-LT002B		"	"	"	"	
		1N34-TC009A	1N34-TCV009A Fails Open	RFPT L.O. Cooler T.C.V. Fails Open	Increased TB CLCW flow cools oil. May create low oil temp alarm	"	"	
		1N34-TC009B	1N34-TCV009B Fails Open	Same	Same	"	"	
1N21-RB-21[3]	N11	1N11-PT004	Loss of Signal	None	None	"	"	
		1N11-PT003	" " "	"	"	"	"	
		1N11-PT011A	" " "	"	"	"	"	
		1N11-PT011B	" " "	"	"	"	"	
		1N11-PT013A	" " "	"	"	"	"	
		1N11-PT013B	" " "	"	"	"	"	
		1N11-PT014A	" " "	"	"	"	"	
		1N11-PT014B	" " "	"	"	"	"	
		1N11-PT014C	" " "	"	"	"	"	
		1N11-PT014D	" " "	"	"	"	"	
		1N11-PT015A	" " "	"	"	"	"	
		1N11-PT015B	" " "	"	"	"	"	
		1N11-PT016A	" " "	"	"	"	"	
		1N11-PT016B	" " "	"	"	"	"	
		1N11-PS017B	None	"	"	"	"	
		1N11-PS015A	None	"	"	"	"	
		1N11-PT018A	Loss of Signal	"	"	"	"	
		1N11-PT018B	" " "	"	"	"	"	
		1N11-PT019	" " "	"	"	"	"	
		1N11-PS019	" " "	"	"	"	"	
		1N11-PS017Y	None	"	"	"	"	
		1N11-PS017Z	None	"	"	"	"	
1N21-RB-21	N21	1N21-PS-001A	Loss of Signal	None Criteria N-2	"	"	"	
		1N21-PS-001B	" " "	" " "	"	"	"	
		1N21-PT-001A	" " "	" " "	"	"	"	
		1N21-PT-001B	" " "	" " "	"	"	"	
		1N21-PT-001	" " "	" " "	"	"	"	
		1N21-PT-004A	" " "	" " "	"	"	"	
		1N21-PT-004B	" " "	" " "	"	"	"	
		1N21-FC010	1N21-FCV010 Fails closed	Water Quality recirc normally closed FCV fails closed	None-Startup System	"	"	

ATTACHMENT 7

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Attachment 7

NOV 24 1982

Docket No.: 50-322

Mr. M. S. Pollock
Vice President - Nuclear
Long Island Lighting Company
175 East Old Country Road
Hicksville, New York 11801

Dear Mr. Pollock:

Subject: Shoreham Nuclear Power Station - Multiple Control System Failure Concern (SER Issue No. 47)

In a letter dated August 27, 1982, (SNRC-761; J. L. Smith to Harold R. Denton) you submitted information to address a control system issue identified in Section 7.7 of the Shoreham Safety Evaluation Report. The staff has conducted a preliminary review of the information submitted and it has been determined that, while your response appears to satisfactorily address the effects of power supply failures, it does not address control system failures caused by common sensors, hydraulic headers, and impulse lines. While the control system issue identified in Section 7.7 does not specifically detail the review of failures caused by hydraulic headers or impulse lines to two or more control systems, informal NRC staff contact with your staff, and the precedent established in the closure of this item on other dockets, has identified these areas of concern. The common sensors concern was identified in Section 7.7. The specific request for information is included in Enclosure 1.

Please inform us, within seven (7) days of receipt of this letter, of your schedule of submittal of the requested information. If you have any questions on this matter, please contact NRC Project Manager, Edward Weinkam at (301) 492-8430.

Sincerely,

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
As stated

cc: See next page

Shoreham

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CONCERN THAT COMMON ELECTRICAL POWER SOURCES OR SENSOR MALFUNCTIONS MAY CAUSE MULTIPLE CONTROL SYSTEM FAILURES

A number of concerns have been expressed regarding the adequacy of safety systems in mitigation of the kinds of control system failures that could actually occur at nuclear plants, as opposed to those analyzed in FSAR Chapter 15 safety analyses. Although the Chapter 15 analyses are based on conservative assumptions regarding failures of single control systems, systematic reviews have not been reported to demonstrate that multiple control system failures beyond the Chapter 15 analyses could not occur because of single events. Among the types of events that could initiate such multiple failures, the most significant are in our judgement those resulting from failure or malfunction of power supplies or sensors common to two or more control systems.

To provide assurance that the design basis event analyses adequately bound multiple control system failures you are requested to provide the following information:

1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

3) Indicate which, if any, of the control systems identified in (1) receive input signals from common sensors, common hydraulic headers, or common impulse lines.

The response should provide justification that simultaneous malfunctions of control systems which could result from failure of a power source, sensor, hydraulic header or sensor impulse line supplying power or signals to more than one control system are bounded by the analysis of anticipated operational ccurrences in Chapter 15 of the Final Safety Analysis Report.

ATTACHMENT 8

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

Direct Dial Number

April 22, 1983 SNRC-872

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SER Issue No. 47 - Control System Failures
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference: (1) Letter SNRC-761 dated 8/27/82
(2) Letter NRC (A. Schwencer) to LILCO (M. S. Pollock) dated 11/24/82

Dear Mr. Denton:

As stated in section 7.7 of Supplement No. 1 to the Shoreham Safety Evaluation Report (SER) the Long Island Lighting Company had committed to conduct a review to demonstrate that failures or malfunctions of power sources or sensors providing power or signals to two or more control systems will not result in consequences outside the bounds of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.

In fulfillment of this commitment, LILCO, via the reference (1) letter, had forwarded a report entitled "Control System Failures Evaluation Report". This report concluded that, although new transient category events can be postulated by considering these failures or malfunctions, the net effects have been positively determined to be less severe than those of the original, conservative Chapter 15 events.

Per the reference (2) letter, the staff conducted a preliminary review of this report and determined that, while the report appeared to satisfactorily address the effects of power supply failures, it did not address control system failures caused by common sensors such as hydraulic headers and impluse lines. In this reference (2) letter, a request for information containing 3 items was included as Enclosure 1. Items 1 and 2 of this request involve the identification of control systems whose failure or malfunction could impact plant safety and further identification of the control systems noted above which receive power from common power sources. LILCO has determined that Sections 3.2 and 3.5 of the Ref. 1 report address these items, and no additional changes are required. Item 3 of this request involves control system failures caused by common sensors, common hydraulic headers or common impluse lines. Control system failures caused by common sensors are enveloped by the Ref. 1 report, and no further revisions are required. This report did not however, address control system failures caused by common hydraulic headers or common impluse lines. LILCO had informed the staff that these failures would be evaluated and any problems would be summarized in a preliminary report, with a final report to be submitted 45 days later.

The purpose of this letter is to serve as the above mentioned preliminary report and advise the staff that an evaluation has been performed as required for control system failures caused by common hydraulic headers or common impluse lines. This evaluation has led to the conclusion that these failures will not result in consequences outside the bounds of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.

A final report is presently being completed and will be submitted to the staff by mid-May. This report will be in the form of a supplement to the report submitted in Ref. 1.

Please advise if you have any questions on this matter.

Very truly yours,

W. J. Museler

for J. L. Smith
Manager, Speical Projects
Shoreham Nuclear Power Station

RWG:bc

cc: J. Higgins
All Parties Listed in Attachment 1

ATTACHMENT 1

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ATTACHMENT 9

LILCO

LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

Direct Dial Number

June 20, 1983 SNRC-905

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SER Issure No. 47 - Control System Failures
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference: (1) Letter SNRC-761 dated 8/27/82
(2) Letter NRC (A. Schwencer) to LILCO (M. S. Pollock dated 11/24/82
(3) Letter SNRC-872 dated 4/22/82

Dear Mr. Denton:

As stated in section 7.7 of Supplement No. 1 to the Shoreham Safety Evaluation Report (SER) the Long Island Lighting Company had committed to conduct a review to demonstrate that failures or malfunctions of power sources or sensors providing power or signals to two or more control systems will not result in consequences outside the bounds of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.

In fulfillment of this commitment, LILCO, via the reference (1) letter, had forwarded a report entitled "Control System Failures Evaluation Report". Per the reference (2) letter, the staff conducted a preliminary review of this report and determined that, while the report appeared to satisfactorily address the effects of power supply failures, it did not address control system failures caused by common sensors such as hydraulic headers and impulse lines.

LILCO subsequently performed and completed an evaluation of control system failures caused by common hydraulic headers or common impulse lines. As stated in Ref. 3, this evaluation concluded that these failures will not result in consequences outside the bounds of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems. At that time, the final report was in the course of preparation.

This report, entitled "Common Sensors Failures Evaluation Report" dated May, 1983, has been completed and forty (40) copies are enclosed herewith for your review.

LILCO believes that the information included herein is sufficient to completely resolve any remaining staff concerns on this issue. Should you have any questions, please contact this office.

Very truly yours,

Original signed by

J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

RWG:bc

Enclosure

cc: J. Higgins
All Parties Listed in Attachment 1

<u>ATTACHMENT 1</u>

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SHOREHAM COMMON SENSORS FAILURES

EVALUATION REPORT

MAY 1983

PREPARED FOR

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

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12-0287
1

CONTENTS

COMMON SENSORS FAILURES EVALUATION REPORT
FOR LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

1.0 OBJECT

This document constitutes:

- An analysis in response to the NRC concern that the failure of an instrument line which contains sensors to multiple control systems could result in consequences outside the bounds of the Shoreham Nuclear Power Station (SNPS) Final Safety Analysis Report (FSAR) Chapter 15 analysis and beyond the capabilities of operator responses or safety systems.

- A positive demonstration that adequate review and analysis has been performed to ensure that, despite such failure, the Shoreham FSAR Chapter 15 analyses are bounding, and no consequence beyond the capability of operator responses or safety systems would result.

A comprehensive approach was developed by General Electric (GE) to analyze the control systems capable of affecting reactor water level, pressure, or power in the SNPS Control Systems Failures Evaluation Report. This report uses the knowledge gained from the Control Systems Failures Evaluation Report for a valid restriction of this analysis to only those systems, which affect reactor water level, pressure or power.

This report was prepared by GE for the Long Island Lighting Company's (LILCO) Shoreham Nuclear Power Station with a significant technical contribution from the Stone & Webster Engineering Corporation (SWEC), the principal architect engineer.

2.0 CONCLUSIONS

This report, supplemented by the existing FSAR Chapter 15 transient analyses, documents an evaluation of the Shoreham Nuclear Power Station for common sensor failures. Evaluation of a broken or plugged instrument standpipe on the feedwater heaters indicated that there would be a reduction in the temperature of the feedwater to the reactor vessel and/or a turbine trip. This combined loss of feedwater heating plus turbine trip transient was not analyzed in Chapter 15 analyses. Subsequent evaluation of reduced feedwater temperature, followed by a turbine trip, indicates the consequences to be bounded by the events considered in the Chapter 15 analysis. All of the analyzed consequences of common instrument failures are bounded by FSAR Chapter 15 analysis.

3.0 ANALYSIS METHODOLOGY

The common sensor failure analysis was conducted in the following manner by GE and SWEC:

Activity	Assigned To
• Identify Common Sensors	SWEC & GE
• Determine Failure Modes	SWEC & GE
• Summarize Common Sensor Failures	GE
• Analyze Combined Effects	GE
• Compare Results to Chapter 15	GE
• Analyze Exceptions	GE
• Modify/Augment Chapter 15 if Necessary	GE

3.1 IDENTIFY COMMON SENSORS

The following systems have been identified as being capable of affecting reactor parameters:

SYSTEMS CAPABLE OF AFFECTING
REACTOR WATER LEVEL, PRESSURE, OR POWER

MPL	Systems
B21	Nuclear Boiler
B31	Reactor Recirculation
C11	CRD Hydraulic
C32	Feedwater
C51	Neutron Monitoring
D11	Process Radiation Monitoring
D21	Area Radiation Monitoring
G33	Reactor Water Cleanup
N11	Exhaust Steam
N21	Condensate and Feedwater
N32	Main Turbine Control
N34	Main Turbine and Feedwater Turbines Lube Oil
N35	Moisture Separator and Heater Drains
N36	Bleed Steam - Extraction Steam
N42	Main Generator Hydrogen and Hydrogen Seal
N43	Main Generator Cooling
N44	Air Removal
N45	Generator Hydrogen and CO_2
N51	Main Generator Excitation
N62	Offgas
N71	Circulating Water
P41	RB Service Water
P42	RB Cooling Water
P43	TB Cooling Water
P50	Compressed Air
P71	Low Conductivity Drains
Z93	Primary Containment Instrumentation

The systems which can affect reactor parameters, as determined by the Control Systems Failures Evaluation Report, were analyzed for multiple system sensors or multiple system contacts from a common instrument line. Instrument lines which serve only one system were eliminated because their failure effects are bounded by the current Chapter 15 analysis.

3.2 DETERMINE FAILURE MODES

An instrument line may fail in the following ways. A broken line will cause pressure instruments to falsely sense a low pressure condition. For a differential pressure instrument a broken reference line will result in a maximum differential pressure condition, while a variable line break will cause a minimum differential pressure condition. Because of the very small line break, the depressurization and loss of coolant effects are negligible for all of these analyses.

A plugged or pinched line will maintain the instrument at the condition it was at the time of failure. To ensure the worst case Chapter 15 consequence is represented the pinched or plugged line will be analyzed at full power. Any line failures which produce instrument conditions differing from those described above are bounded by those conditions produced by the completely broken or completely plugged cases.

3.3 SUMMARIZE COMMON SENSOR FAILURES

The common sensor failure table (attached) lists the results of this investigation. The table identifies all instruments which are connected to a particular line, their failure modes, and their direct effect on the reactor.

3.4 ANALYZE COMBINED EFFECTS

This step totaled all of the individual effects of each instrument failure. The interaction of each effect relative to one another was evaluated and combined effect consequences were determined and are described in Section 4.0, "Common Sensor Summary Results and Chapter 15 Comparison."

3.5 COMPARE RESULTS TO CHAPTER 15

The combined effects, as identified in Section 4.0, were compared to the existing Chapter 15 analysis to determine if any new transient was discovered which is not bounded by the current analysis.

3.6 ANALYZE EXCEPTIONS

There were no exceptions to FSAR Chapter 15 analysis.

3.7 MODIFY/AUGMENT CHAPTER 15 IF NECESSARY

This step was not necessary in the Shoreham analysis.

4.0 COMMON SENSOR SUMMARY RESULTS AND CHAPTER 15 COMPARISONS

Instrument Line*	Line Failure Consequences
No. 1	None
No. 2	A break of this line will cause immediate reduction of feedwater flow. At worst, this will be identical to a loss of feedwater event as described in the Chapter 15 analysis. A plugged line can cause, at worst, a loss of feedwater event or feedwater controller maximum demand event. Both of these events are considered in the Chapter 15 analysis.
No. 3	A break in this line will immediately initiate a reactor scram and cause an increase in feedwater flow. Reactor scram takes precedence over other transients and is considered in the Chapter 15 analysis. A plugged line will produce the same consequences as described for Line No. 2.
No. 4	None
No. 5	A broken instrument line will initiate a main and feedwater turbine trip. This sequence is similar to the loss of feedwater event described for Instrument Line No. 2. A plugged line will produce the same consequences as described for Instrument Line No. 2.
No. 6	A broken instrument line will cause an immediate increase in feedwater flow and also disable all high level turbine trip logic. This event is described in Chapter 15 analysis and is similar to the failure of the feedwater controller maximum demand with an additional single failure. A plugged line will produce the same consequences as described for Instrument Line No. 2.
Nos. 7,9,11,13	A break in any of these lines will cause feedwater flow to decrease. This will ultimately lead to a reactor scram on low reactor water level. This event is bounded by the analyzed loss of feedwater event described in the Chapter 15 analysis.

*See lower left corner of each table page.

Instrument Line	Line Failure Consequences
	A plugged line will cause an error in feedwater steam flow following resulting in, at worst, a gradual increase or decrease in reactor water level. Both events are bounded by the Chapter 15 analysis.
Nos. 8,10,12,14	A break in any of these lines initiates an immediate MSIV closure and increase in reactor feedwater flow. The MSIV closure event will take precedence, causing a reactor scram as the MSIVs begin to close. This event is considered in Chapter 15 analysis.
	A plugged line will produce the same consequences as described for Instrument Line No. 7.
Nos. 15,16	A break in one of these lines initiates rod block. Rod block will not cause adverse consequences.
	A plugged line causes no consequences that affect reactor water level, pressure or power.
Nos. 17,19,21,23,25	A break in any of these lines would result in a reduction in temperature of feedwater to the reactor vessel. This event is bounded by Chapter 15 analysis.
	A plugged line could cause either a false high or false low water level signal in the heater or reheater. This would result in a reduction in temperature of feedwater to the reactor vessel and/or a turbine trip. The feedwater temperature could drop by an estimated 20°F. Subsequent evaluation of reduced feedwater temperature, followed by a turbine trip, indicate the consequences to be bounded by the events considered in Chapter 15 analysis.
Nos. 18,20,22,24,26	A break in any of these lines would result in a reduction in temperature of feedwater to the reactor vessel and/or a turbine trip. Subsequent evaluation of reduced feedwater temperature, followed by a turbine trip, indicate the consequences to be bounded by the events considered in Chapter 15 analysis.
	A plugged line could cause either a false high or false low water level signal in the heater or reheater. This would result in reduction in temperature of feedwater to the reactor vessel and/or a turbine trip. The feedwater temperature will drop by an estimated 20°F. Subsequent evaluation of reduced feedwater temperature, followed by a turbine trip, indicate the conseqences to be bounded by the events considered in Chapter 15 analysis.

Instrument Line	Line Failure Consequences
No. 27	A break of this line would possibly result in isolation of the Reactor Water Cleanup System. This will not affect reactor water level, pressure, or power. A plugged line causes no consequences that affect reactor water level, pressure, or power.
No. 28	A break in this line causes no consequences that affect reactor water level, pressure, or power. A plugged line causes no consequences that affect reactor water level, pressure, or power.

SHOREHAM COMMON SENSOR FAILURE TABLE

TABLE PAGE 1

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
NUCLEAR BOILER	B21-N027	BROKEN	MAXIMUM REACTOR VESSEL DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	B21-R605 METER INOPERATIVE	NONE
		PLUGGED	INACCURATE REACTOR VESSEL DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	B21-R605 METER AT INACCURATE READING	NONE
FEEDWATER	C32-N017	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (WIDE RANGE-UPSET REACTOR LEVEL)	C32-R608 RECORDER INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (WIDE RANGE-UPSET REACTOR LEVEL)	C32-R608 RECORDER AT INACCURATE READING	NONE

INSTRUMENT LINE 1 3/4"-K-22-1502-2 "B" SIDE REFERENCE LEG

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
NUCLEAR BOILER	B21-N020C AND B21-N020D	BROKEN	NONE; PRESSURE SWITCH NO LONGER IN RPS CIRCUIT, ELECTRICALLY DISABLED	NONE	NONE
		PLUGGED	NONE; PRESSURE SWITCH NO LONGER IN RPS CIRCUIT, ELECTRICALLY DISABLED	NONE	NONE
	B21-N097C AND B21-N097D	BROKEN	MINIMUM PRESSURE SIGNAL	RHR LOW PRESSURE PERMISSIVE SYSTEM A AND SYSTEM B; ATWS & RPT C&D HIGH PRESSURE TRIP DISABLED; ½ RECIRC DISCHARGE VALVE CLOSURE PERMISSIVE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	½ RHR LOW PRESSURE SYSTEM A AND SYSTEM B TRIP INOPERATIVE; ATWS/RPT C&D HIGH PRESSURE TRIP DISABLED; LOSS OF RECIRC DISCHARGE VALVE CLOSURE PERMISSIVE	NONE
	B21D-N078C AND B21-N078D	BROKEN	MINIMUM PRESSURE SIGNAL	HIGH PRESSURE SCRAM TRIP FOR CHANNELS "A2" AND "B2" DISABLED; "A1" AND "B1" BACKUP AVAILABLE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	HIGH PRESSURE SCRAM TRIP FOR CHANNELS "A2" AND "B2" DISABLED; "A1" AND "B1" BACKUP AVAILABLE	NONE

INSTRUMENT LINE 2 3/4"-K-24-1502-2 "B" SIDE REFERENCE LEG

PAGE 1 OF 4

12-0285 (2)

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
MSIV-LCS	E32-N050 AND E32-N060	BROKEN	MINIMUM PRESSURE SIGNAL	MAIN STEAM LINE PRESSURE BELOW SETPOINT; MSIV-LCS INBOARD SYSTEM PERMISSIVE FOR MANUAL INITIATION AND METER E32-R660 INOPERATIVE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	MAIN STEAM LINE PRESSURE ABOVE SETPOINT; MSIV-LCS INBOARD SYSTEM INOPERATIVE AND METER E32-R660 AT	NONE
NUCLEAR BOILER	B21-N080C AND B21-N080D	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	WATER LEVEL 3 SCRAM CHANNELS "A2" AND "B2" INOPERATIVE; "A1" AND "B1" BACKUP AVAILABLE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	WATER LEVEL 3 SCRAM CHANNELS "A2" AND "B2" INOPERATIVE; "A1" AND "B1" BACKUP AVAILABLE	NONE
NUCLEAR BOILER	B21-N055B	BROKEN	MINIMUM PRESSURE SIGNAL	LEVEL/PRESSURE RECORDER B21-R623B INOPERATIVE; RECORDER AT HIGH SPEED	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	LEVEL/PRESSURE RECORDER B21-R623B AT CONSTANT READING	NONE
	B21-N095B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	ADS (B) WATER LEVEL 3 PERMISSIVE LOST.	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	ADS (B) WATER LEVEL 3 PERMISSIVE LOST.	NONE

INSTRUMENT LINE 2 3/4"-K-24-1502-2 "B" SIDE REFERENCE LEG

SHOREHAM COMMON SENSOR FAILURE TABLE

TABLE PAGE 4

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N004B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606B WATER LEVEL RECORDER INOPERATIVE; REACTOR FEEDWATER DECREASED FLOW	SIMILAR TO LOSS OF FEEDWATER EVENT, LEADS TO SCRAM, MSIV CLOSURE, HPCI/RCIC INITIATION
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606B INACCURATE READING; REACTOR FEEDWATER ERROR IN LEVEL FOLLOWING	GRADUAL INCREASE OR REDUCTION OF LEVEL IN REACTOR.
NUCLEAR BOILER	B21-N091B AND B21-N091D	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	CORE SPRAY SYSTEM B INITIATION INOPERATIVE; RHR SYSTEM B INITIATION INOPERATIVE; ATWS-ARI SYSTEM B INITIATION INOPERATIVE; ATWS-RPT SYSTEM B INITIATION INOPERATIVE; ½ HPCI/RCIC TURBINE TRIP SIGNAL.	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	CORE SPRAY SYSTEM B INITIATION INOPERATIVE; RHR SYSTEM B INITIATION INOPERATIVE; ATWS-ARI SYSTEM B INITIATION INOPERATIVE; ATWS-RPT SYSTEM B INITIATION INOPERATIVE; HPCI INJECTION SHUTOFF INOPERATIVE	NONE
NUCLEAR BOILER	B21-N081C AND B21-N081D	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	B21-R623B AND B21-R604 INOPERATIVE HALF OF WATER LEVEL 2 MAIN STEAM LINE ISOLATION INOPERATIVE	NONE

INSTRUMENT LINE 2 3/4"-K-24-1502-2 "B" SIDE REFERENCE LEG

PAGE 3 OF 4

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	HALF OF WATER LEVEL 2 MAIN STEAM LINE ISOLATION INOPERATIVE	NONE
NUCLEAR BOILER	B21-N037B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL	B21-R610 LEVEL INDICATOR INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL	B21-R610 LEVEL INDICATION INACCURATE READING	NONE
NUCLEAR BOILER	B21-N026D	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C61-LIT007 LOCAL LEVEL INDICATOR INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C61-LIT007 LOCAL LEVEL INDICATOR INACCURATE READING	NONE
FEEDWATER	C32-N008	BROKEN	MINIMUM PRESSURE SIGNAL	C32-R609 RECORDER REACTOR PRESSURE PEN INOPERATIVE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	C32-R609 RECORDER REACTOR PRESSURE PEN AT CONSTANT READING	NONE

INSTRUMENT LINE 2 3/4"-K-24-1502-2 "B" SIDE REFERENCE LEG

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
NUCLEAR BOILER	B21-N080C AND B21-N080D	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	WATER LEVEL 3 SCRAM	REACTOR SCRAM
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	WATER LEVEL 3 CHANNELS "A2" AND "B2" INOPERATIVE	NONE
	B21-N095B	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	REACTOR LOW WATER SYSTEM B LAMP LIGHTS. ADS B LOW LEVEL PERMISSIVE (OTHER PERMISSIVES REQUIRED)	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	ADS B WATER LEVEL 3 PERMISSIVE LOST	NONE
FEEDWATER	C32-N004B	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606B WATER LEVEL RECORDER INOPERATIVE; REACTOR FEEDWATER INCREASED FLOW; & TURBINE TRIP	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606B WATER LEVEL RECORDER AT INACCUARTE READING; REACTOR FEED-WATER ERROR IN LEVEL FOLLOWING	GRADUAL INCREASE OR REDUCTION IN REACTOR WATER LEVEL

INSTRUMENT LINE 3 3/4"-K-25-1502-2 "B" SIDE VARIABLE LEG

SHOREHAM COMMON SENSOR FAILURE TABLE

TABLE PAGE 7

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
	C32-N017	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (WIDE RANGE-UPSET REACTOR LEVEL)	C32-R608 RECORDER INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (WIDE RANGE-UPSET REACTOR LEVEL)	C32-R608 RECORDER AT INACCURATE WIDE RANGE-UPSET REACTOR LEVEL SIGNAL	NONE
	B21-N027	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (REACTOR VESSEL LEVEL)	B21-R605 METER INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (REACTOR VESSEL LEVEL)	B21-R605 METER AT INACCURATE READING	NONE

INSTRUMENT LINE 3 3/4"-K-24-1502-2 "B" SIDE VARIABLE
PAGE 2 OF 2

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
NUCLEAR BOILER	B21-N032	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL	B21-R613 RECORDER CORE PLATE DIFFERENTIAL PRESSURE BLACK PEN INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL	B21-R613 RECORDER CORE PLATE DIFFERENTIAL PRESSURE BLACK PEN AT INACCURATE READING	NONE
CRD HYDRAULICS	C11-N008	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE	C11-R009 LOCAL AND CONTROL ROOM C11-R602 METERS INOPERATIVE	NONE
		PLUGGED	IN[illegible]ATE DIFFERENTIAL PRESSURE	C11-R009 LOCAL AND CONTROL ROOM C11-R602 METERS AT INACCURATE READING	NONE
	C11-N011	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE	C11-R005 LOCAL AND C11-R603 COOLING WATER METERS INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE	C11-R005 LOCAL AND C11-R603 COOLING WATER METER AT INACCURATE READING	NONE
CORE SPRAY	E21-N004A	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL	SYSTEM I HIGH DIFFERENTIAL PRESSURE SPRAY HEADER TO TOP OF CORE PLATE ANNUNCIATION	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL	NONE	NONE

INSTRUMENT LINE 4   3/4"-K-55-1502-2   PRESSURE ABOVE CORE PLATE

PAGE 1 OF 2

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
	E21-N004B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL	SYSTEM 2 HIGH DIFFERENTIAL PRESSURE SPRAY HEADER TO TOP OF CORE PLATE ANNUNCIATION	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL	NONE	NONE

INSTRUMENT LINE 4 3/4"-K-55-1502-2 PRESSURE ABOVE CORE PLATE

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
NUCLEAR BOILER	B21-N078A AND B21-N078B	BROKEN	MINIMUM PRESSURE SIGNAL	HIGH VESSEL PRESSURE SCRAM TRIP FOR CHANNELS "A1" AND "B1" DISABLED; "A2" AND "B2" BACKUP AVAILABLE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	HIGH VESSEL PRESSURE SCRAM TRIP FOR CHANNELS "A1" AND "B1" DISABLED; "A2" AND "B2" BACKUP AVAILABLE	NONE
	B21-N020A AND B21-N020B	BROKEN	GIVES MSIV CLOSURE BYPASS SIGNAL TO SCRAM TRIP LOGIC (A_1 AND B_1) ONLY IN SHUTDOWN, REFUELING AND STARTUP MODE	NONE - ELECTRICALLY DISABLED	NONE
		PLUGGED	REACTOR HIGH PRESSURE TRIP OF MSIV CLOSURE SCRAM BYPASS IS INOPERATIVE	NONE - ELECTRICALLY DISABLED	NONE
	B21-N097A AND B21-N097B	BROKEN	MINIMUM PRESSURE SIGNAL	RHR LOW PRESSURE PERMISSIVE SYSTEM A AND SYSTEM B; ATWS AND RPT A&B HIGH PRESSURE TRIP DISABLED; ½ RECIRC DISCHARGE VALVE CLOSURE PERMISSIVE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	½ RHR LOW PRESSURE SYSTEM A AND SYSTEM B TRIP INOPERATIVE; ATWS AND RPT A AND B PRESSURE TRIP DISABLED; LOSS OF ½ RECIRC DISCHARGE VALVE CLOSURE PERMISSIVE.	NONE

INSTRUMENT LINE 5 3/4"-K-28-1502-2 "A" SIDE REFERENCE LEG

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
NUCLEAR BOILER	B21-N080A AND B21-N080B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	WATER LEVEL 3 TRIP CHANNELS "A1" AND "B1" INOPERATIVE: "A2" AND "B2" BACKUP AVAILABLE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	WATER LEVEL 3 TRIP CHANNELS "A1" AND "B1" INOPERATIVE; "A2" AND "B2" BACKUP AVAILABLE	NONE
	B21-N095A	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	ADS (A) WATER LEVEL 3 PERMISSIVE LOST.	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL	ADS (A) WATER LEVEL 3 PERMISSIVE LOST.	NONE
FEEDWATER	C32-N004A	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606A WATER LEVEL RECORDER INOPERATIVE; REACTOR FEEDWATER DECREASED FLOW; ½ MAIN AND FEEDWATER TURBINE TRIP ON HIGH WATER LEVEL.	C32-N004A AND C FAILURE HIGH, MAIN AND FEEDWATER TURBINE TRIP DUE TO HIGH SENSED LEVEL BY A AND C INSTRUMENTS, FEEDWATER FLOW LOST.
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606A WATER LEVEL RECORDER AT INACCURATE READING; REACTOR FEED-WATER ERROR IN LEVEL FOLLOWING	GRADUAL INCREASE OR REDUCTION IN REACTOR WATER LEVEL

INSTRUMENT LINE 5 3/4"-K-28-1502-2 "A" SIDE REFERENCE LEG

PAGE 2 OF 5

12-0265 (11)

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
	C32-N004C	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606C WATER LEVEL RECORDER INOPERATIVE; & MAIN TURBINE AND FEEDWATER TRIP ON HIGH WATER LEVEL	SEE C23-N004A
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606C WATER LEVEL RECORDER AT INACCURATE READING.	NONE
	C32-N005	BROKEN	MINIMUM PRESSURE SIGNAL	C32-R605 REACTOR HIGH PRESSURE RECORDER INOPERATIVE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	C32-R605 REACTOR HIGH PRESSURE RECORDER AT CONSTANT READING	NONE
NUCLEAR BOILER	B21-N055A	BROKEN	MINIMUM PRESSURE SIGNAL	LEVEL/PRESSURE RECORDER B21-R623A, LEVEL/PRESSURE METERS 21-L1004 AND B21-P1004 INOPERATIVE; RECORDER AT HIGH SPEED	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	LEVEL/PRESSURE RECORDER B21-R623A, LEVEL/PRESSURE METERS B21-L1004 AND B21-P1004 AT CONSTANT READING	NONE
NUCLEAR BOILER	B21-N091A AND B21-N091C	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	CORE SPRAY SYSTEM A INITIATION INOPERATIVE; RHR SYSTEM A INITIATION INOPERATIVE; ATWS-ARI SYSTEM A INITIATION INOPERATIVE; ATWS-RPT SYSTEM A INITIATION INOPERATIVE; & RCIC/HPCI TURBINE TRIP SIGNAL.	NONE

INSTRUMENT LINE 5 3/4"-K-28-1502-2 "A" SIDE REFERENCE LEG

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	CORE SPRAY SYSTEM A INITIATION INOPERATIVE; RHR SYSTEM A INITIATION INOPERATIVE; ATWS-ARI SYSTEM A INITIATION INOPERATIVE; ATWS-RPT SYSTEM A INITIATION INOPERATIVE; RCIC TURBINE STOP INOPERATIVE; HPCI INJECTION SHUTOFF INOPERATIVE.	NONE
	B21-N081A AND B21-N081B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	B21-R23A RECORDER INOPERATIVE HALF OF WATER LEVEL 2 MAIN STEAM LINE ISOLATION INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	HALF OF WATER LEVEL 2 MAIN STEAM LINE ISOLATION INOPERATIVE	NONE
REMOTE SHUTDOWN	C61-N006	BROKEN	MINIMUM PRESSURE SIGNAL	C61-R011 PRESSURE INDICATOR INOPERATIVE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	C61-R011 PRESSURE INDICATOR AT CONSTANT READING	NONE
NUCLEAR BOILER	B21-N026B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C61-R010 LEVEL INDICATOR INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C61-R010 LEVEL INDICATOR AT INACCURATE READING	NONE

INSTRUMENT LINE 5　　3/4"-K-28-1502-2　　"A" SIDE REFERENCE LEG

12-0265 (13)

SHOREHAM COMMON SENSOR FAIL'RE TABLE

TABLE PAGE 14

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
	B21-N037A	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	B21-R615 REACTOR WATER LEVEL RECORDER INOPERATIVE	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	B21-R615 REACTOR WATER LEVEL RECORDER AT INACCURATE READING	NONE

INSTRUMENT LINE 5 3/4"-K-28-1502-2 "A" SIDE REFERENCE LEG
PAGE 5 OF 5

12-0285 (14)

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
NUCLEAR BOILER	B21-N080A AND B21-N080B	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	WATER LEVEL 3 SCRAM	REACTOR SCRAM
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	WATER LEVEL 3 TRIP CHANNELS "A1" AND "B1" INOPERATIVE	NONE
	B21-N095A	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	REACTOR LOW WATER SYSTEM A LAMP LIGHTS. ADS LOW WATER LEVEL 3 PERMISSIVE (OTHER PERMISSIVES REQUIRED)	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	ADS LOW WATER LEVEL 3 PERMISSIVE LOST	NONE
	C32-N004A	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606A WATER LEVEL RECORDER INOPERATIVE; REACTOR FEEDWATER INCREASED FLOW, LOSS OF TURBINE TRIP LOGIC.	C32-N004A AND C BROKEN, WILL CAUSE FEEDWATER TO FILL VESSEL PAST L8 TRIP, FLOOD STEAMLINES. TURBINE TRIP LOGIC LOST ON HIGH WATER LEVEL. TURBINE TRIP ON HIGH VIBRATION.
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606A WATER LEVEL RECORDER AT INACCURATE READING; REACTOR FEED-WATER ERROR IN LEVEL FOLLOWING	GRADUAL INCREASE OR REDUCTION IN REACTOR WATER LEVEL.
	C32-N004C	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606C WATER LEVEL RECORDER INOPERATIVE; LOSS OF TURBINE TRIP LOGIC.	SEE C32-N004A

INSTRUMENT LINE 6 3/4"-K-29-1502-2 "A" SIDE VARIABLE LEG

PAGE 1 OF 2

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (LEVEL)	C32-R606C WATER LEVEL RECORDER AT INACCURATE READING; LOSS OF TURBINE TRIP LOGIC.	NONE
MSIV-LCS	E32-N058	BROKEN	MINIMUM PRESSURE SIGNAL	LOW PRESSURE PERMISSIVE OF OUTBOARD MSIV LEAKAGE CONTROL SYSTEM AND METER E32-R658 OPERATIVE	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	OUTBOARD MSIV LEAKAGE CONTROL SYSTEM INOPERATIVE AND METER E32-R658 AT CONSTANT READING	NONE

INSTRUMENT LINE 6 3/4"-K-29-1502-2 "A" SIDE VARIABLE LEG

SHOREHAM COMMON SENSOR FAILURE TABLE

TABLE PAGE 17

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N003A	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (FLOW)	REACTOR FEEDWATER DECREASED FLOW	FEEDWATER DECREASED FLOW
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (FLOW)	REACTOR FEEDWATER ERROR IN STEAM FLOW FOLLOWING	NONE
NUCLEAR BOILER	B21-N006A AND B21-N006B	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE

INSTRUMENT LINE 7
PAGE 1 OF 1

3/4"-K-1-1502-2 3/4"-K-101-1502-2
MAIN STEAM LINE FLOW INSTRUMENTATION LINES

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N003A	BROKEN	MAXIMUM DIFFE ENTIAL PRESSURE (FLOW)	REACTOR FEEDWATER INCREASED FLOW	FEEDWATER INCREASED FLOW
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE (FLOW)	REACTOR FEEDWATER ERROR IN STEAM FLOW FOLLOWING	NONE
NUCLEAR BOILER	B21-N006A AND B21-N006B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE	MAIN STEAM ISOLATION VALVES CLOSURE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE

INSTRUMENT LINE 8
PAGE 1 OF 1

3/4"-K-2-1502-2 3/4"-K-102-1502-2
MAIN STEAM LINE FLOW INSTRUMENTATION LINES

12-0285 (18)

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N003B	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (FLOW)	REACTOR FEEDWATER DECREASED FLOW	FEEDWATER DECREASED FLOW
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (FLOW)	REACTOR FEEDWATER ERROR IN STEAM FLOW FOLLOWING	NONE
NUCLEAR BOILER	B21-N007A AND B21-N007B	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE

INSTRUMENT LINE 9
PAGE 1 OF 1

3/4"-K-5-1502-2 3/4"-K-105-1502-2
MAIN STEAM LINE FLOW INSTRUMENTATION LINES

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N003B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE (FLOW)	REACTOR FEEDWATER INCREASED FLOW	FEEDWATER INCREASED FLOW
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE (FLOW)	REACTOR FEEDWATER ERROR IN STEAM FLOW FOLLOWING	NONE
NUCLEAR BOILER	B21-N007A AND B21-N00.3	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE	MAIN STEAM ISOLATION VALVES CLOSURE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE

INSTRUMENT LINE 10
PAGE 1 OF 1

3/4"-K-6-1502-2 3/4"-K-106-1502-2
MAIN STEAM LINE FLOW INSTRUMENTATION LINES

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N003C	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (FLOW)	REACTOR FEEDWATER DECREASED FLOW	FEEDWATER DECREASED FLOW
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (FLOW)	REACTOR FEEDWATER ERROR IN STEAM FLOW FOLLOWING	NONE
NUCLEAR BOILER	B21-N006A AND B21-N006B	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE

INSTRUMENT LINE 11 3/4"-K-9-1502-2 3/4"-K-109-1502-2
PAGE 1 OF 1 MAIN STEAM LINE FLOW INSTRUMENTATION LINES

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6″

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6″

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N003C	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE (FLOW)	REACTOR FEEDWATER INCREASED FLOW	FEEDWATER INCREASED FLOW
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE (FLOW)	REACTOR FEEDWATER ERROR IN STEAM FLOW FOLLOWING	NONE
NUCLEAR BOILER	B21-N008A AND B21-N008B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE	MAIN STEAM ISOLATION VALVES CLOSURE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE

INSTRUMENT LINE 12
PAGE 1 OF 1

3/4"-X-110-1502-2 3/4"-X-10-1502-2
MAIN STEAM LINE FLOW INSTRUMENTATION LINES

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N003D	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (FLOW)	REACTOR FEEDWATER DECREASED FLOW	FEEDWATER DECREASED FLOW
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (FLOW)	REACTOR FEEDWATER ERROR IN STEAM FLOW FOLLOWING	NONE
NUCLEAR BOILER	B21-N009A AND B21-N009B	BROKEN	MINIMUM DIFFERENTIAL PRESSURE SIGNAL (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE SIGNAL (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE

INSTRUMENT LINE 13 3/4"-K-13-1502-2 3/4"-K-113-1502-2
PAGE 1 OF 1 MAIN STEAM LINE FLOW INSTRUMENTATION LINES

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
FEEDWATER	C32-N003D	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE (FLOW)	REACTOR FEEDWATER INCREASED FLOW	FEEDWATER INCREASED FLOW
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE (FLOW)	REACTOR FEEDWATER ERROR IN STEAM FLOW FOLLOWING	NONE
NUCLEAR BOILER	B21-N009A AND B21-N009B	BROKEN	MAXIMUM DIFFERENTIAL PRESSURE (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE	MAIN STEAM ISOLATION VALVES CLOSURE
		PLUGGED	INACCURATE DIFFERENTIAL PRESSURE (FLOW)	MAIN STEAM ISOLATION VALVES CLOSURE CHANNELS A&B INOPERATIVE; BACKED UP BY CHANNELS C&D	NONE

INSTRUMENT LINE 14
PAGE 1 OF 1

3/4"-K-14-1502-2 3/4"-K-114-1502-2
MAIN STEAM LINE FLOW INSTRUMENTATION LINES

SHOREHAM COMMON SENSOR FAILURE TABLE

SYSTEM ID	COMMON TAP SENSOR MPL	FAILURE TYPE (BROKEN OR PLUGGED)	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECT
CRD HYDRAULICS	C11-N054A	BROKEN	MINIMUM PRESSURE SIGNAL	OPEN CONTACT TO ROD SEQUENCE CONTROL SYSTEM, CAUSING ROD BLOCK	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	MAINTAIN CLOSED CONTACT TO ROD SEQUENCE CONTROL SYSTEM	NONE
MAIN STEAM	1N11-PT003	BROKEN	MINIMUM PRESSURE SIGNAL	1ST STAGE TURBINE PRESSURE RECORDER 1N11-PR003 INOPERATIVE.	NONE
		PLUGGED	CONSTANT PRESSURE SIGNAL	1ST STAGE TURBINE PRESSURE RECORDER 1N11-PR003 AT CONSTANT READING	NONE

INSTRUMENT LINE 15   1ST STAGE TURBINE PRESSURE INSTRUMENT LINE

ATTACHMENT 10

Attachment 10

NUREG-0420
Supplement No. 4

Safety Evaluation Report
related to the operation of Shoreham Nuclear Power Station, Unit No. 1

Docket No. 50-322

Long Island Lighting Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1983

7 INSTRUMENTATION AND CONTROLS

7.4 Systems Required for Safe Shutdown

7.4.3 Remote Shutdown System

On the basis of its review of the information furnished by the applicant regarding the remote shutdown panel (RSP) as reported in Section 7.4.3 of SSER 3, the NRC staff found that the design of the RSP would meet the regulatory requirements specified in GDC 19 and the guidance as detailed in the SRP Sections 7.4 II and III. As a confirmatory item, the staff required the applicant to provide final operating procedures and Technical Specifications and also perform a system operational verification test of the RSP with the assumption of the most limiting single failure in the equipment train controlled from the RSP or remote stations away from the RSP.

In a letter dated June 21, 1983, from J. L. Smith to Harold R. Denton, the applicant committed to (1) conduct a walk-through prior to fuel load to demonstrate RSP system operability (including stations remote from the RSP) with the assumption of the most limiting single failure, (2) revise the operating procedures prior to exceeding 5% power to reflect the final design of the RSP and its remote stations, and (3) address the RSP and its remote stations in the Shoreham Technical Specifications.

The staff has concluded that the above commitments are acceptable and that this confirmatory item is resolved.

However, the staff will condition the Shoreham license to require the applicant to (1) implement (and document) all of the required design changes discussed in Section 7.4.3 of SSER 3 by the end of the first refueling and (2) perform an acceptable procedure verification test for the new RSP design at that time.

7.5 Safety-Related Display Instrumentation

In SER Section 7.5, the NRC staff requested that the applicant review the adequacy of emergency operational procedures used by control room operators to attain safe shutdown upon loss of any Class 1E or non-Class 1E buses supplying power to safety- or nonsafety-related instruments and to control systems. The response to this request addressed Items 1 and 3 of IEB 79-27 regarding plant system design features. Based on a protection sequence for shutdown developed for Shoreham, the applicant demonstrated that only Class 1E systems are necessary to achieve cold shutdown and, therefore, enough equipment would remain available after the loss of any Class 1E or non-Class 1E electrical bus. This conclusion was accepted by the NRC staff in SSER 1.

In addition, the applicant committed to conduct a failure mode effects analysis of plant electrical buses and to determine whether emergency operating procedures are adequate for dealing with the resultant plant conditions. The applicant submitted the results of this analysis in letter, SNRC-761, dated August 27, 1982. The analysis demonstrated that failures or malfunctions of power sources or sensors providing power or signals to two or more control systems will not result in consequences outside the bounds of the FSAR Chapter 15 accident analysis. Plant personnel used the bus tree and load tables developed in the control systems failure analysis to verify that the plant operating procedures were adequate to deal with the identified transients. No procedure changes were required.

Finally, the applicant committed to review the Shoreham alarm response procedures for loss of power to Class 1E buses to ensure that these procedures identify the indications and symptoms resulting from postulated power failures on 4-kV, 480-V, and 125-V dc buses. The applicant concluded that the station operating procedures were adequate to address loss of power conditions on any Class 1E bus. CILAR B79-27 was logged closed for this item on September 29, 1982.

An NRC inspector reviewed the station normal operating, abnormal operating, and alarm response procedures for the 4-kV, 480-V, 125-V dc, 120-V ac instrument, and 120-V ac uninterruptible power supply systems as part of an inspection documented in Inspection Report No. 50-322/83-02. This review determined that the procedures provide sufficient, detailed instructions for the operator to: (1) identify the alarms, indicators, and symptoms needed to diagnose a loss of bus power; (2) restore bus power; and (3) identify alternate indications that may be used for plant control.

The NRC staff, therefore considers this item to be resolved.

7.6 Other Instrumentation and Control Systems Required for Safety

7.6.6 Physical Independence

7.6.6.1 Physical Independence Within NSSS Cabinets

During the NRC staff preparations for the Shoreham hearings, a concern developed regarding the lack of physical separation between non-Class 1E and Class 1E circuits inside the NSSS cabinets at Shoreham. It appeared to the NRC staff that the design of the Shoreham electrical system failed to provide adequate physical independence of circuits inside the NSSS cabinets, as established in current regulatory practice.

Section 4.6 of Standard 279-1971 of the Institute of Electrical and Electronics Engineers (IEEE), "Criteria for Protection Systems for Nuclear Power Generating Stations," requires, in part, that channels that provide signals for the same protective functions be independent and physically separated. Regulatory Guide (RG) 1.75, "Physical Independence of Electric Systems," describes a method acceptable to the NRC staff for complying with IEEE 279-1971 with respect to physical independence of the circuits and electrical equipment comprising or associated with the Class 1E power system, the protection system, systems actuated or controlled by the protection systems, and auxiliary or supporting systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions.

In addition, in accordance with Section 4.6 of IEEE 384-1974, "IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits" (endorsed

7.6.6.2 Electrical Separation Barriers

Deficiencies in separation for Shoreham electrical cables and raceways were identified in IE Inspection Report 50-322/79-07 and subsequent reports. As a result, the NRC staff required each deficiency to be corrected using one of the following four options:

(1) Correct the deficiency by meeting the electrical equipment separation criteria set forth in FSAR Section 3.12.

(2) Correct the deficiency by meeting RG 1.75, Revision 2, dated September 1978.

(3) Correct the deficiency by installing an acceptable barrier.

(4) Justify the deficiency by performing a specific analysis for each cable or raceway where the minimum separation is not met to demonstrate that a failure will not propagate because of the insufficient separation.

With regard to Option 3, the applicant, by letter dated January 14, 1983, provided its definition and basis (substantiated by test) for what constitutes an acceptable barrier. The applicant defined an acceptable barrier as a single conduit, tray cover, or wrapping of Siltemp woven-ceramic tape with 3M Scotch Branch No. 69 glass tape.

The NRC staff has reviewed Wyle Test Report No. 46287, "Test Report on Thermal Barrier and Short Circuit Test on 600 VAC Power and 120 VAC Control Cables," and Engineering and Design Coordination Report F-41238K, which describes the separation guidelines to be used for the installation of barriers at Shoreham. Based on the NRC staff's review of these reports, on discussions with the applicant, and on the conservatism of the proposed design, the NRC staff concludes that the applicant's definition of an acceptable barrier meets the objectives of IEEE 384-1974, as augmented by RG 1.75, and meets the independence requirements of GDC 17. It is, therefore, acceptable.

7.7 Control Systems Not Required for Safety

7.7.1 High-Energy Line Breaks (IE Bulletin 79-22, "Qualification of Control System")

If control systems are exposed to the environment resulting from the rupture of reactor coolant lines, steamlines, or feedwater lines, the control systems may malfunction in a manner that would cause consequences to be more severe than assumed in safety analyses.

The NRC staff requested the applicant to perform a review to determine what, if any, design changes or operator actions would be necessary to ensure that high-energy line breaks (HELBs) would not cause control system malfunctions and complicate the event beyond the FSAR analysis. In response to this concern, the applicant initiated a review to determine whether HELBs could have an effect on multiple control systems and to investigate the impact of failure of the applicable systems on the FSAR Chapter 15 analysis.

By letter dated November 8, 1982, from J. L. Smith to H. R. Denton (NRC), the applicant provided a report that presented the results of a design review, evaluation and plant walkdown addressing this concern.

The procedure that the applicant followed to perform the HELB analysis is as follows. The applicant

(1) Identified nonsafety-related control systems and componants within these systems that may impact reactor pressure, water level, or critical power ratio and that may be vulnerable to functional damage from HELBs

(2) Established the assumptions and resulting criteria for high-energy line determination, break postulation, and consequence evaluation.

(3) Identified the locations (elevations/areas) that contain high-energy piping systems and in which components for the nonsafety-related control systems are located.

(4) Conducted a walkdown of the areas to verify the location of nonsafety-related control components and determined their proximity to high-energy lines.

(5) Postulated breaks in the areas having components from one or more of these nonsafety-related control systems and determined the resultant effect on the components, and ultimately the controlled equipment. Areas having no multiple system interactions within the constraints of the above criteria were not considered.

(6) Determined the resultant state of the reactor as a result of simultaneous failure of these nonsafety-related control systems.

(7) Compared this to events already analyzed and reported in FSAR Chapter 15, and determined if they are bounded. If not bounded, additional analysis was performed to determine if the effects are significant.

(8) Identified HELB/nonsafety-related control system events that were determined to be significant based on this analysis and indicated the corrective action to be taken.

The applicant performed the HELB study using the guidelines noted above. The results of the study indicated that all postulated events satisfy the criteria for infrequent events, i.e., that the dose consequences do not exceed 10% of the 10 CFR 100 criteria.

The most limiting event was found to be the loss of feedwater heating exacerbated by a turbine trip. This condition could be caused by a pipe break within the turbine building, which may simultaneously cause a partial loss of feedwater heating and a turbine trip, if the appropriate controls are disabled, leading to improper valve positions.

The loss of feedwater heating would cause a gradual increase in reactor power level which, without operator action, could eventually lead to a reactor trip at the APRM trip setpoint (117% power). Depending upon the specific timing of the event, the turbine trip may occur at a reactor power elevated between the operating value and the trip level of 117%.

When the turbine trip takes place, the bypass valves would start dumping 25% of the main steam flow to the condenser until the condenser pressure reaches 22.5 inches Hga. At this time, the bypass would also trip shut automatically. The bypass would be in operation for approximately 7 seconds after the turbine trip.

The staff was concerned about the consequences of an assumed worst case single failure concurrent with any of the postulated HELB events being more severe than those of the FSAR Chapter 15 analyses. The applicant provided the results of an analysis using the single failure assumption for the postulated worst case scenarios. The worst case postulated single failure analyzed was the complete loss of the turbine bypass system concurrent with the most limiting event noted above. This analysis shows that the results are well within the criteria for infrequent events. The applicant stated in a letter dated August 2, 1983 (from J. L. Smith to Harold R. Denton) that the dose consequences for this worst-case event will not exceed a small fraction (<10%) of the 10 CFR 100 criteria.

The staff questioned the applicant regarding the effects of humidity, pressure, and temperature on system components as a result of the HELB. In a letter dated May 11, 1983, from J. L. Smith to Harold R. Denton, the applicant stated that the effects of humidity, pressure, and temperature on the operability of these nonsafety-related control systems was addressed in formulating the conclusions reached in the original report. For additional clarification, the applicant stated that for small confined zones, it was assumed that any HELB would affect all nonsafety-related control components within the zone. Using this approach, it is apparent that the environmental effects on these components are directly enveloped within the scope of the report. In large, more open zones, only the components within the range of high-energy lines were assumed to fail simultaneously with the pipe break. Environmental effects on components outside the range of these HELB large open areas would tend to develop relatively slowly in comparison to the dynamic effects that would lead to rapid automatic and operator-initiated mitigative actions.

Based on its review and the conclusions of the applicant's study that indicate that the dose consequences will not exceed 10% of 10 CFR 100 criteria, the staff finds that SER Open Item 48, "High Energy Line Breaks," is resolved.

7.7.2 Multiple Control System Failures

SSER 3 noted that the applicant had committed to conduct a review to identify any power sources or sensors that provide power or signals to two or more control systems and to demonstrate that failures or malfunctions of these power sources or sensors will not result in consequences beyond the bounds of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.

By letter dated August 27, 1982, the applicant submitted a control systems failures evaluation report. The review performed for this report used the event-consequence logic of the Chapter 15 analyses, but started the logic chain from the specific source (i.e., a single bus failure) rather than a system condition.

This approach uncovered previously unanalyzed interactions. Although these new transient category events were postulated as a result of this study, it was concluded that the net effects were less severe than those of the original FSAR Chapter 15 events. The results of this report demonstrated that the previously reported limits of minimum critical power ratio, peak vessel, and main steamline pressures, and peak fuel cladding temperature for the expected operational occurrence category of events would not be exceeded as a result of common power source or sensor failures.

However, the staff remained concerned about control system malfunctions caused by a single failure of common hydraulic headers or impulse lines. The applicant submitted a report (letter dated June 20, 1983, from J. L. Smith to Harold R. Denton) addressing this issue. This report, supplemented by the existing FSAR Chapter 15 transient analysis, documents an evaluation of the Shoreham design related to postulated common sensor line failures (i.e., common hydraulic headers, impulse lines). Failures of common hydraulic headers, sensor taps, and instrument lines feeding two or more control system inputs were identified. Failure modes (broken or plugged lines were postulated for 28 individual identifications) and the resulting effects were analyzed.

All of the consequences of common instrument line failures were bounded by the previous analyses presented in FSAR Chapter 15, with the exception of a broken or plugged instrument standpipe on the feedwater heaters, which would reduce the feedwater temperature going into the reactor vessel and result in a possible turbine trip. Subsequent evaluation of this event indicates that the consequences are, in fact, bounded by the events considered in the Chapter 15 analyses.

The staff has reviewed the bases and results for the applicant's study and concludes, with reasonable assurance, that the consequences of single failures within the control systems are bounded by the analyses in FSAR Chapter 15. Therefore, the staff has concluded that SER Open Item 47, "Multiple Control System Failures," is resolved.

ATTACHMENT 11

LILCO
LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

November 8, 1982

SNRC-786

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SER Issue No. 48, High Energy Line Breaks
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Dear Mr. Denton:

As stated in section 7.7 of Supplement No. 1 to the Shoreham Safety Evaluation Report (SER), the Long Island Lighting Company committed to conduct a review to demonstrate that the harsh environments associated with high energy line breaks do not cause control system malfunctions and result in consequences more severe than those of the Chapter 15 analyses or beyond the capability of operators or safety systems.

In fulfillment of this commitment, enclosed are forty (40) copies of a report entitled, "High Energy Line Break/Control System Failure Analysis". This report presents the results of a comprehensive study, including a walkdown of the plant areas, that was conducted (1) to identify non-safety control systems and components that may be affected by postulated pipe breaks, and then (2) to conservatively determine the state of the reactor as a result of the simultaneous failure of all affected non-safety control systems. It is concluded that all conditions resulting from the postulated pipe break events (10 conditions resulting from a postulated pipe break were evaluated individually and in a combination) are bounded by the Chapter 15 analysis. With the exception of the loss of feedwater heating exacerbated by a turbine trip at elevated reactor power levels, the transient events meet the conservative limits of the transient category. It is concluded, based upon analyses, inspection procedures, and operator action, that this event is a low-frequency accident event which is bounded by the accident events of the FSAR Chapter 15. Therefore, the postulated high energy line break, with resulting adverse affects of non-safety control systems, poses no significant risk to the health and safety of the public. No further accident analysis or any design modification is necessary.

The submittal of this report completes the confirmatory information required by the Staff to completely close out SER Issue Number 48.

Should you have any questions, please contact this office.

Very truly yours,

J. L. Smith

J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

RJT:mp

Enclosure

cc: J. Higgins
All parties

ATTACHMENT 12

Attachment 12

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 24 1983

Docket No.: 50-322

Mr. M. S. Pollock
Vice President - Nuclear
Long Island Lighting Company
175 East Old Country Road
Hicksville, New York 11801

Dear Mr. Pollock:

Subject: Request for Additional Information Regarding High Energy Line Breaks

The staff is continuing its evaluation of your November 8, 1982, submittal (SNRC-786) regarding SER open item #48 - Effects of High Energy Line Breaks on Control Systems for the Shoreham Nuclear Power Station. In order to complete that review, we need your response to the questions in the enclosure to this letter within 30 days of your receipt of this letter.

If you have any questions please contact Ralph Caruso, the licensing project manager, (301) 492-9793.

Sincerely,

A. Schwencer /for

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
As stated

cc: See next page

Shoreham

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ADDITIONAL INFORMATION REQUIRED FOR STAFF
EVALUATION OF THE EFFECTS OF HIGH ENERGY
LINE BREAKS ON CONTROL SYSTEMS

If Control Systems are exposed to the environment resulting from the rupture of reactor coolant lines, steam lines or feedwater lines, the control system may malfunction in a manner which would cause consequences to be more severe than assumed in safety analyses.

The staff requested a review to determine what, if any, design changes or operator actions would be necessary to assure high energy line breaks would not cause control system malfunctions and complicate the event beyond the FSAR analysis.

By letter dated November 8, 1982, (J. L. Smith to Harold R. Denton) the Shoreham applicant submitted information summarizing the results of a design review, evaluation and plant walkdown addressing this concern.

Our review of the effects of high energy line breaks on control systems cannot be fully completed until additional information as discussed below is provided by the applicant.

(1) the results of analysis of the effects of humidity, pressure, and temperature, in addition to the effects of pipe whip and jet impingement, on the operability of control systems.

(2) Clarification of the single failure assumption used in the study. The intent of this concern was to have the applicant review the possibility of consequential control system failures which exacerbate the effects of high energy line breaks and

take action where needed, to assure that the postulated events would be adequately mitigated. In conjunction with the above (high energy line break and consequential control system failures), an additional single failure within the systems used to mitigate this event should be considered. This assumption concerning the additional single failure is standard regulatory practice and is also discussed in IEEE 379-1979, "IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems." This standard basically states that adequate protective action must be provided to accomplish a protective function in the presence of any single detectable failure concurrent with all identifiable but non-detectable failures, all failures occurring as a result of the single failure, and all failures which would be caused by the design basis event requiring the protective function.

ATTACHMENT 13

Attachment 13

LILCO

LONG ISLAND LIGHTING COMPANY

SHOREHAM NUCLEAR POWER STATION

P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

Direct Dial Number

May 11, 1983

SNRC-887

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SER Issue No. 48 - High Energy Line Breaks
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Reference: (1) Letter SNRC-786 dated 11/8/82
(2) Letter NRC (A. Schwencer) to LILCO (M. S. Pollock) dated 1/24/83

Dear Mr. Denton:

In response to SER Issue No. 48, "High Energy Line Breaks" (HELB), LILCO had submitted the reference (1) letter forwarding a report entitled "High Energy Line Break/Control System Failure Analysis". This report represented a comprehensive study, including a walk-down of plant areas, that was conducted (1) to identify non-safety control systems and components that may be affected by postulated pipe breaks and then (2) to conservatively determine the state of the reactor as a result of the simultaneous failure of all affected non-safety control systems. It was concluded that all conditions resulting from the postulated pipe break events are bounded by the accident analyses contained in Chapter 15 of the FSAR, and are therefore capable of being mitigated either automatically or by operator action.

In the reference (2) letter, the staff advised that their review of the above noted report cannot be fully completed until LILCO provides additional information on the effects of humidity, pressure and temperature on the operability of these non-safety control systems.

These effects have been addressed in formulating the conclusions reached in the HELB report, although a brief clarification may be beneficial. As stated in Section 4.1 "Analysis Methodology" two general methods were used to analyze the pipe break zones utilized in the study. For small confined zones, it was assumed that any HELB would incapacitate all non-safety control components within the zone. This assumption was made even though specific components may not be affected by the jet impingement or pipe whip resulting from a specific break. Using this conservative "sarificial approach", it becomes apparent that the environmental effects on these components are directly enveloped within the scope of the report.

In large, more open zones, only the components within the range of the high energy lines were assumed to fail simultaneously with the pipe break. This is consistent with the goals of the study, to determine whether the result of FSAR Chapter 15 accident analyses are exceeded. FSAR Chapter 15 analyses primarily address short term effects where limiting values generally occur very rapidly after event initiation. Assuming a reactor scram, automatic actions would quickly take place to mitigate the immediate effects of the event. Environmental effects on components in these large spaces would tend to develop relatively slowly in comparison to the dynamic effects on the components which would lead to more rapid automatic and operator initiated mitigative actions.

In addition, the staff requested, in the reference (2) letter, that the HELB study consider an additional single failure within the systems used to mitigate the event. In response, two examples of postulated worst-case scenarios were evaluated for the Shoreham plant. These two scenarios are identified below:

<u>CASE I</u>

a) HELB occurs in Turbine Building
b) Loss of feedwater heating occurs, causing reactor power increase to 117% of rated.
c) Turbine generator trip occurs coincident with peak reactor power
d) Scram occurs as a result of turbine generator trip. Loss of offsite power also occurs.
e) HPCI fails (Single failure)
f) RCIC operates
g) Reactor water level is restored by RCIC.

<u>CASE II</u>

a) Steps a through d are the same as CASE I
e) Loss of turbine bypass to condenser (single failure)
f HPCI operates
g Reactor water level is restored by HPCI

The occurrence of these events is extremely unlikely. This conclusion is based on consideration of the probability that a combination of the worst case conditions occurs concurrently:

- The worst case pipe segment breaks on the most important line;
- HELB can affect all controllers in an area and cause failures in worst case modes;
- Breaks occur at worst case locations (in reality, many of these locations have low calculated stress levels and thus are unlikely to fracture);
- Both turbine trip and reactor high power-level trip occur at appropriate (i.e. worst cases) times;
- Additional single failure occurs

Regardless, these two cases were analyzed quantitatively using conservative Chapter 15 analysis models for the two analyses and the results indicate that the short term part of the event with bypass (turbine trip at the thermal power monitor setpoint power) is enveloped by the FSAR Chapter 15 Accident Analysis. In this case, the peak fuel cladding temperature is less than 900°F as compared to 2200°F limit. The second event which imposed a failure of the turbine bypass system on the initial scenarios was estimated to reach a peak cladding temperature of about 1200°F, again well within the FSAR Chapter 15 Accident limits. This further confirms the conclusions outlined in the reference (1) letter.

It should be noted that the long term plant cooldown of these two events with various system failures, such as HPCI inoperative, are addressed in the Emergency Procedure Guidelines developed for these types of concerns.

The submittal of this information should be sufficient to close SER issue No. 48.

Should you have any further questions, please contact this office.

Very truly yours,

Original signed by

J. L. Smith
Manager, Special Projects
Shoreham Nuclear Power Station

[illegible]

[illegible]
All Parties Listed in Attachment 1

ATTACHMENT 1

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ATTACHMENT 14