CONTROL SYSTEMS FAILURES
EVALUATION REPORT

SEPTEMBER 1982

PREPARED

FOR

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
WNP-2 NUCLEAR POWER STATION
(HANFORD 2)

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CONTROL SYSTEMS FAILURES
EVALUATION REPORT
FOR WNP-2 NUCLEAR POWER STATION

1.0 OBJECT

This document constitutes:

- An analysis in response to the NRC concern that the failures of power sources which provide power or electrical signals to multiple control systems could result in consequences outside the bounds of the WNP-2 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 failures the WNP-2 FSAR Chapter 15 analyses are bounding, and no consequences 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 WNP-2 plant.

This report with its attachments was prepared by the General Electric Company (GE) for the Washington Public Power Supply System (WPPSS). Significant technical contribution was provided from Burns & Roe, Inc (BRI).

2.0 CONCLUSIONS

This report, supplemented by the existing FSAR Chapter 15 analyses, documents an evaluation of the WNP-2 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 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 division of responsibility in performing this analysis is as listed below:

TASKS	ASSIGNED TO
• DEFINE BUS STRUCTURE	BRI
• DEFINE CONTROL SYSTEMS	BRI & GE
• IDENTIFY LOADS & EFFECTS DUE TO BUS LOSS	BRI & GE
• DETERMINE CRITICAL LOADS	BRI & GE
• SUMMARIZE CRITICAL LOADS	GE
• ANALYZE COMBINED EFFECTS	BRI & 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 Figure 1) 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 WNP-2 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 BRI, is as follows:

3.2 DEFINE CONTROL SYSTEMS (Continued)

SYSTEMS	SYSTEMS
Feedwater Control System	Hydrogen Seal System
Nuclear Boiler System	Generator Cooling
Reactor Recirculation	Air Removal
CRD Hydraulic Control System	Generator Hydrogen & CO2 Purge
Feedwater Turbine	Main Generator Excitation
Neutron Monitoring	Off Gas
Process Radiation Monitor System	Circulating Water
Area Radiation Monitor System	Service Water
Reactor Water Cleanup	RB Closed Cooling Water System
Main Steam	TB Closed Cooling Water System
Condensate	Compressed Air
Turbine Control	Low Conductivity Drains
Lube Oil P/O TG, RFW	Primary Containment
Moisture Extraction	Instrumentation
Exhaust Steam	Heater Drains
Bleed (Extraction) Steam	Miscellaneous Drains
Heater Vents	Sealing Steam
Feedwater	Plant Service Water
Service Air	

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 BRI, 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 system 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. Combined effects at intermediate bus levels which were more severe than their associated higher bus combined failures were analyzed and included in section 4.

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 WNP-2 analysis.

4.0 BUS LOSS SUMMARY RESULTS AND CHAPTER 15 COMPARISON

4.1 AC BUS

4.1.1 SM-1 (4.16KV)

The loss of this bus causes reactor feedwater pump 1A, condensate pump 1A, condensate booster pump 2A, and circulating water pump 1A to be inoperative, and the air removal system to be isolated.

There is a partial loss in feedwater heating since the extraction steam motor-operated valves to the feedwater heaters fail open (as-is) on loss of motive power, but steam is partially bypassed to the condenser via the dump valves which also fail open.

There is also a slow loss in the main condenser vacuum due to the isolation of the air removal system and loss of one circulating water pump.

Concurrent with the feedwater pump 1A trip and a reactor vessel low water level, the reactor recirculation flow runback reduces reactor power to about 68% NBR to stay within the remaining feedwater capacity and avoid scram.

If no operator action were taken, a main turbine trip due to low condenser vacuum would occur more than ninety minutes after the power bus loss and at a reduced reactor power level.

This event is bounded by the turbine trip already analyzed in FSAR Chapter 15 and bounded by the loss of lower bus, PP-1B-A.

4.1.1.1 PP-1B-A (120V)

The loss of this bus causes the reactor feedwater pump turbine 1A to be inoperative. There is also a partial loss of feedwater heating, and a slow loss of main condenser vacuum due to the isolation of the air removal system, leading to a main turbine trip.

The worst case reduction in feedwater temperature has been determined to be considerably less than 83°F.

Concurrent with the feedwater pump trip and a low reactor vessel water level, the reactor recirculation flow runback occurs which is intended to reduce reactor power to about 68% NBR. (Within the remaining feedwater capability). The partial reduction of feedwater heating will gradually raise power to about 75%. If the power exceeds feedwater capability, low level scram will occur and this event is bounded by the loss of all feedwater event already analysed in Chapter #15.

Due to the inaccessibility during power operation of the valve that isolates the air removal system the main condenser vacuum loss continues until the main turbine trips and the reactor shuts down.

If no operator action were taken, a main turbine trip due to low condenser vacuum would occur more than a hundred minutes after the power bus loss and at a reduced reactor power level.

This event is bounded by the turbine trip transient already analyzed in FSAR Chapter 15.

4.1.2 SM-2 (4.16KV)

The loss of this bus causes reactor feedwater pump 1B, condensate pump 1B, condensate booster pump 2B, and circulating water pump 1B to be inoperative.

There is also a partial loss of feedwater heating due to the extraction steam to the heaters partially bypassed to the main condenser, and a possible slow loss of main condenser vacuum due to the loss of ejector/condenser "B", leading to a main turbine trip. If ejector/condenser "A" is in use at the time, or if it is manually started, the loss of ejector/condenser "B" would not result in a main turbine trip.

The worst case reduction in feedwater temperature has been calculated to be less than 33°F.

Concurrent to reactor feedwater pump turbine 1A trip and low reactor vessel water level, the reactor recirculation flow runback is initiated which is intended to reduce reactor power to about 68% NBR. The effect of the slightly colder feedwater is to raise this final power slightly but probably still within the feedwater capability.

In the unlikely event of a main turbine trip (more than a hundred minutes after the power bus loss, and at reduced reactor power), the effects of this event are similar to those of the loss of SM-1.

4.1.2.1 PP-2P-A The loss of this bus causes a trip of feedwater pump 1B, a partial loss of feedwater heating, a possible slow loss of main condenser vacuum, due to the loss of ejector/condenser "B". A potential delayed (~100 min.) main turbine trip due to low condenser vacuum.

The worst case reduction in feedwater temperature has been calculated to be considerably less than 83°F. Concurrent with the feedwater pump trip and a low reactor vessel water level, the reactor recirculation flow run back will reduce reactor power to 68% NBR (which is within the remaining feedwater capability).

The partial reduction of feedwater heating will gradually raise power to about 75%. If the power exceeds feedwater capability. Low level scram will occur and this event is bounded by the loss of all feedwater event already analysed in Chapter 15.

Since the reactor is operating at reduced power, should a main turbine trip occur, the event is bounded by the turbine trip transient already analyzed in FSAR Chapter 15.

4.1.3 SM-3

The loss of this bus causes condensate pump 1C, condensate booster pump 2C, and circulating water pump 1C to be inoperative. There is also a partial loss of feedwater heating and a potential delayed (>90 min.) main turbine trip due to low condenser vacuum.

The worst case reduction in feedwater heating is less severe than that due to the loss of power at a lower bus because the motor-operated extraction steam valves fail-open on loss of motive power; this reduction in temperature has been calculated to be no more than 23°F.

The condensate pump 1C trip causes the remaining condensate booster pumps 2A and 2B to trip on low suction head, and causes both feedwater pumps to trip.

The reactor scrams on low water level. The scram occurs rapidly and precedes the main turbine trip and the effects of the loss of feedwater heating on the reactor.

This event is bounded by the loss of feedwater transient already analyzed in FSAR Chapter 15 and by the loss of the lower bus, PP-3A-A.

4.1.3.1 PP-3A-A

The loss of this bus causes a partial loss of feedwater heating, a loss of the main turbine oil temperature control valve, and a potential main turbine trip due to vibration.

The worst case reduction in feedwater temperature has been calculated to be no more than 51°F. This reduction in feedwater heating will increase reactor power by less than eleven percent nuclear boiler rated (NBR) power.

The worst case scenario is the unlikely event of a loss of feedwater heating and a delayed main turbine trip.

A computer analysis was performed to determine the reactor parameters as a consequence of a main turbine trip at approximately 111 percent of initial steady state power (turbine trip at ~116 NBR power). The results yielded a delta critical power ratio (ΔCPR) of less than 0.15, and a maximum vessel pressure of 1177 psia which are less severe than the consequences of the loss of feedwater heater, manual flow control, and the feedwater controller failure-max. demand at high power transients analyzed in FSAR Chapter 15. This event is then, although previously not analyzed for the WNP-2 plant, still bounded by existing analyses.

4.2 DC BUS

4.2.1 S1-7

The loss of this bus causes a trip of both reactor feedwater pump turbines and a potential delayed trip of the main turbine.

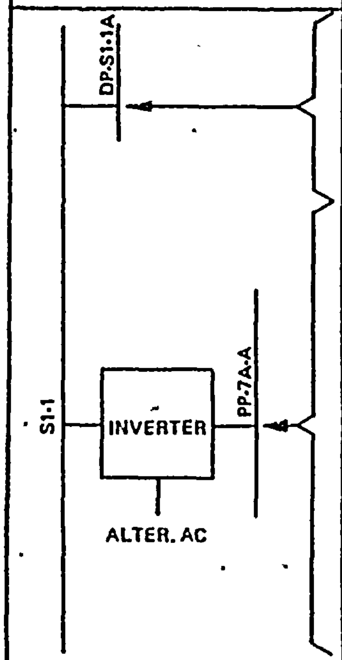
Following the trip of both feedwater turbines, the reactor vessel water level lowers and the reactor scrams on the low water level.

The scram occurs rapidly and precedes the would-be main turbine trip.

This event is bounded by the loss of feedwater transient already analyzed in FSAR Chapter 15.

APPENDIX A

HANFORD CONTROL SYSTEM FAILURE ANALYSIS



DC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
	REACTOR FEEDWATER	REACTOR VESSEL HIGH WATER LEVEL, RFP TRIP CHANNEL "C"	CHANNEL "C" TRIPS ON LOSS OF POWER	NONE, HIGH WATER LEVEL TRIP FUNCTION REQUIRES 2 OF 3 CHANNELS TO TRIP RFPT'S AND MAIN TURBINE.	NONE, CHANNEL "C" INDICATES TRIPPED.
	CONDENSER	COND-PCV-5 VALVE	COND-PCV-5 OPENS	RFW FLOW REDUCED 5000 GPM, DECREASE IN CONDENSER VACUUM	RPV WATER LEVEL 3 SCRAM
	REACTOR FEEDWATER	RELAY TT-X-1A RFPT TRIP INTERLOCK TO REACTOR RECIRC SYSTEM	RELAY TT-X-1A DEENERGIZES PROVIDING RFPT TRIP SIGNAL TO SYSTEM	NONE - BOTH RFPT TRIP AND RPV LOW WATER LEVEL SIGNALS REQUIRED FOR RECIRC RUNBACK	
	STEAM LEAK DETECTION	TEMPERATURE SWITCHES E31-N604AB, N815AB	DISABLE HIGH TEMPERATURE INPUT TO MSIV TRIP	NONE	
	REACTOR FEEDWATER	RFW-FCV-2A	RFW-FCV-2A FULL OPEN	RPV WATER LEVEL 3 SCRAM	

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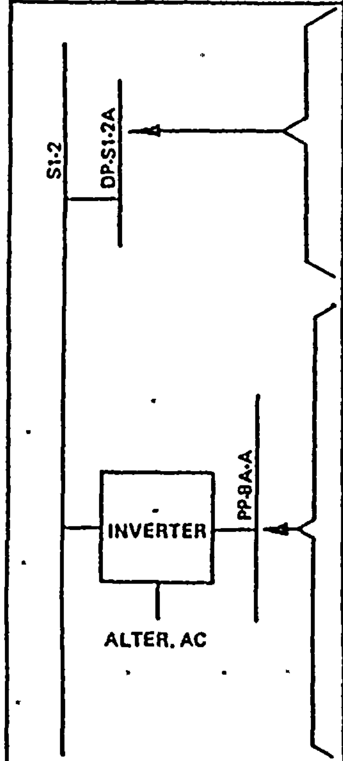

DC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
DP-S1-2A	REACTOR FEEDWATER	REACTOR VESSEL HIGH WATER LEVEL, RFP TRIP CHANNEL "B"	CHANNEL "B" TRIPS ON LOSS OF POWER.	NONE, HIGH WATER LEVEL TRIP FUNCTION REQUIRES 2 OF 3 CHANNELS TO TRIP RFPT'S AND MAIN TURBINE	NONE, CHANNEL "B" INDICATES TRIPPED
PP-8A-A	REACTOR FEEDWATER	RFW–FCV-2B CIRCUIT	FCV-2B OPENS	SCRAM AT RPV LOW LEVEL 3	REACTOR SCRAM AT RPV LOW LEVEL 3
		RFW LOCKUP CIRCUIT	STOP RFP'S IF CONTROL SIGNAL LOST	SCRAM AT RPV LOW LEVEL 3 IF RFP CONTROL SIGNAL LOST	
		RELAY TT-X-1B, RFPT TRIP INTERLOCK TO REACTOR RECIRC. SYSTEM	RELAY TT-X-1B DEENERGIZES PROVIDING RFPT TRIP SIGNAL TO REACTOR RECIRC. SYSTEM	NONE-BOTH RFPT TRIP & RPV LOW WATER LEVEL SIGNALS REQUIRED FOR RECIRC. RUN BACK	
	STEAM LEAK DETECTION	TEMPERATURE SWITCHES E31-N615CD, N604CD	DISABLE HIGH TEMPERATURE INPUT TO MSIV TRIP	NONE	NONE

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DC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
S1-7 DP-S1-1C	REACTOR FEEDWATER	RFP TURBINE "A" SOLENOID TRIP CIRCUIT	RFP TURBINE "A" TRIP	RPV WATER LEVEL LOWER AND REACTOR RECIRC RUN-BACK TO 68% POWER	RPV WATER LEVEL LOWER AND REACTOR RECIRC RUN-BACK TO 68% POWER
DP-S1-1B	AIR REMOVAL	AR-V-1 VALVE AR-V-2 VALVE	LOSE CONDENSER VACUUM SLOWLY	MAIN TURBINE TRIP ≈ 100 MINUTES	MAIN TURBINE TRIP ON LOW CONDENSER VACUUM ≈ 100 MINUTES
DP-S1-2C	REACTOR FEEDWATER	RFP TURBINE "B" SOLENOID TRIP CIRCUIT	RFP TURBINE "B" TRIP	RPV WATER LEVEL LOWER AND REACTOR RECIRC RUN-BACK TO 68% POWER	RPV WATER LEVEL LOWER AND REACTOR RECIRC RUN-BACK TO 68% POWER
	MAIN TURBINE CONTROL	VOLTAGE REGULATOR CONTROL CIRCUIT	MAIN TURBINE TRIP ON LARGE LOAD CHANGE	NO LOAD FOLLOWING	

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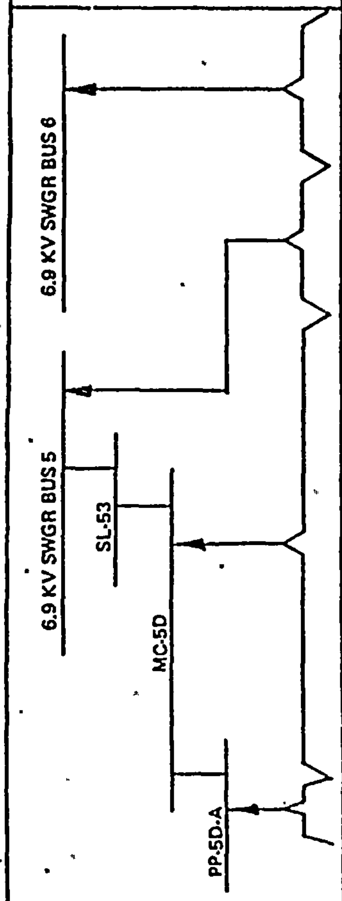

AC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
	REACTOR RECIRC	RECIRC PUMP C001B	PUMP C001B TRIP TO LOW SPEED	REACTOR LOW POWER LEVEL	LOW POWER LEVEL
	REACTOR RECIRC	RECIRC PUMP C001A	PUMP C001A TRIP TO LOW SPEED	REACTOR LOW POWER LEVEL	LOW POWER LEVEL
	OFFGAS	F051A VALVE F051B VALVE	LOSE CONDENSER VACUUM SLOWLY,	MAIN TURBINE TRIP $\approx$ 100 MINUTES	MAIN TURBINE TRIP ON LOW CONDENSER VACUUM $\approx$ 100 MINUTES
	OFFGAS	REGEN. BLOWERS	LOSE CONDENSER VACUUM SLOWLY	MAIN TURBINE TRIP $\approx$ 100 MINUTES	MAIN TURBINE TRIP ON LOW CONDENSER VACUUM $\approx$ 100 MINUTES

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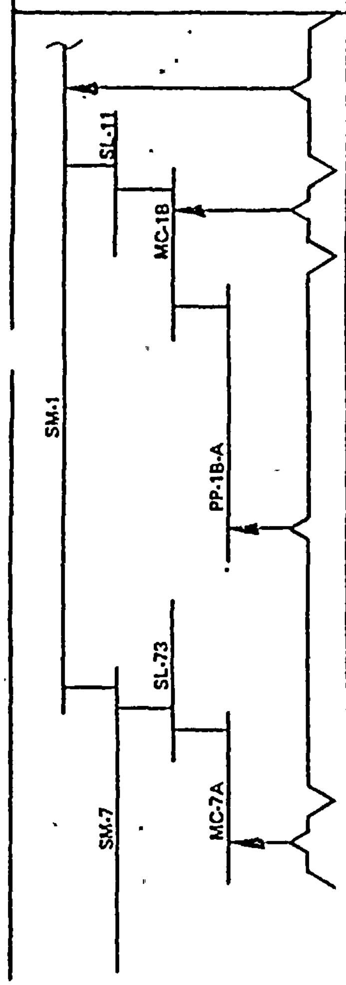

AC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
	REACTOR FEEDWATER	COND. PUMP 1A COND. BOOSTER PUMP 2A	PUMPS INOPERATIVE	DECREASE IN FLOW TO SUCTION OF RFP'S	SEE SECTION 4.0
	CIRC WATER	CIRC. WATER PUMP 1A	PUMP INOPERATIVE	DECREASE IN COOLING FLOW TO MAIN CONDENSER	
	REACTOR RECIRC	LFMG SET S001A	LFMG S001A INOPERATIVE	NONE AT FULL POWER	
	REACTOR FEEDWATER	RFP-TNG-1A (TURNING GEAR) RFP-MOP-1A (MAIN OIL PUMP) RFP-AOP-1A (AUX OIL PUMP)	HOT TNG - RFP 1A TRIP STOP MOP - RFP 1A TRIP STOP AOP - RFP 1A TRIP	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER
	REACTOR FEEDWATER	991X-A (TURBINE SPEED SW)	RFP - 1A TRIP	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER
	AIR REMOVAL	AR-SPV-2A AR-SPV-2B	LOW CONDENSER VACUUM DUE TO AIR EJECTOR "A" LOSS	MAIN TURBINE TRIP IN 103 MINUTES	DECREASE FEEDWATER TEMPERATURE DECREASE CONDENSER VACUUM MAIN TURBINE TRIP IN 103 MINUTES
		AR-SPV-1-1 AR-SPV-1-2	LOW CONDENSER VACUUM DUE TO AIR REMOVAL SYSTEM ISOLATED	MAIN TURBINE TRIP IN 103 MINUTES	
	BLEED STEAM	BS-V-39A VALVE BS-V-35A VALVE BS-V-64 VALVE	BS-V-39A OPEN BS-V-35A OPEN BS-V-64 OPEN	NONE	
	HEATER DRAINS	BS-V-4A VALVE BS-V-5A VALVE BS-V-6A VALVE	BS-V-4A CLOSES BS-V-5A CLOSES BS-V-6A CLOSES	DECREASE IN FEEDWATER TEMPERATURE AND CONDENSER VACUUM	
	HEATER VENTS	HV-V-20A VALVE	HV-V-20A OPENS	NONE	
	SEALING STEAM	SS-V-12A VALVE	SS-V-12A OPENS	NONE	
	OFFGAS	OG-V-129A VALVE	OG-V-129A OPENS	NONE	
	COMPRESSED AIR	INSTR. AIR COMPRESSOR 1A COMP. 1B ON BUS SM-3, COMP. 1C ON BUS SM-2	COMPRESSOR 1A INOPERATIVE	NONE - COMPRESSORS 1B & 1C MEET REQUIREMENTS	NONE

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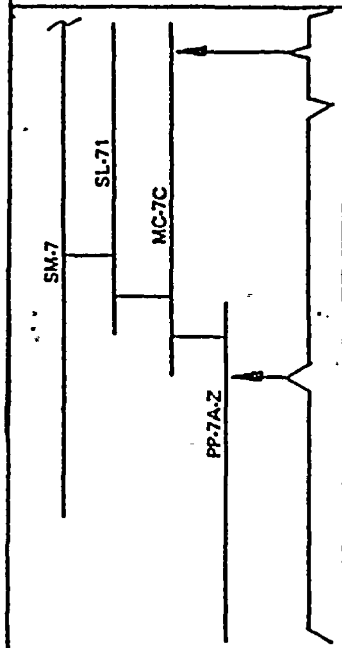
HANFORD CONTROL SYSTEM FAILURE ANALYSIS



AC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
	REACTOR RECIRC	SUBLOOP HYD. PWR UNIT D003A (SEE MC-3D-A)	IF OTHER LOOP NOT AVAILABLE FCV 60A LOCKS UP	NO LOAD FOLLOWING	NO LOAD FOLLOWING
	REACTOR FEEDWATER	RFPT-GOV-1A	RFW PUMP 1A INOPERATIVE	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER
		FEEDWATER SYSTEM CONTROL CIRCUITRY	FEEDWATER PUMPS AT LAST SPEED	NO LOAD FOLLOWING WITH REDUCED POWER	RPV HIGH WATER LEVEL TRIP
		REACTOR VESSEL HIGH WATER LEVEL, RFP TRIP CHANNEL "A"	CHANNEL "A" TRIPS ON LOSS OF POWER	NONE, HIGH LEVEL TRIP FUNCTION REQUIRES 2 OF 3 CHANNELS TO TRIP RFPT'S AND MAIN TURBINE	CHANNEL "A" INDICATES TRIPPED
	REACTOR RECIRC	LOOP A & B FLOW CONTROLLERS	LOCKUP OF FC VALVES 60A & 60B	NO LOAD FOLLOWING	
		FLUX CONTROLLER FLUX ESTIMATOR	SHIFT CONTROL TO MANUAL MODE	MANUAL LOAD FOLLOWING	

APPENDIX A

HANFORD CONTROL SYSTEM FAILURE ANALYSIS

AC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
SM-2	REACTOR FEEDWATER	COND. PUMP 1B	PUMPS INOPERATIVE	DECREASE IN FLOW TO SUCTION OF RFP'S	SEE SECTION 4.0
		COND. BOOSTER PUMP 2B			
	CIRC. WATER	CIRC. WATER PUMP 1B		DECREASE IN CONDENSER COOLING FLOW	
MC-2P	COMPRESSED AIR	INSTR. AIR COMPRESSOR 1C	COMPRESSOR 1C INOPERATIVE (COMP. 1A ON BUS SM-1, COMP. 1B ON BUS SM-3)	NONE, COMPRESSORS 1A & 1B MEET REQUIREMENTS	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68 % POWER
	REACTOR FEEDWATER	RFP-TNG-1B (TURNING GEAR) RFP-MOP-1B (MAIN OIL PUMP) RFP-AOP-1B (AUX OIL PUMP)	HOT TNG-RFP1B TRIP STOP MOP - RFP 1B TRIP STOP AOP - RFP 1B TRIP	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER	
SL-21 PP-2P-A	REACTOR FEEDWATER	(TURBINE SPEED SW) 99 T1X-1B	RFP-1B TRIP	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER
	MAIN STEAM	MS-V-142B VALVE	MS-V-142B OPENS	NONE	DECREASE IN FEEDWATER FLOW
	BLEED STEAM	BS-V-44B VALVE BS-V-45B VALVE BS-V-115B VALVE	BS-V-44B OPENS BS-V-45B OPENS BS-V-115B OPENS	NONE	DECREASE IN FEEDWATER TEMPERATURE DECREASE IN CONDENSER VACUUM
	HEATER VENTS	HV-V-29B VALVE	HV-V-29B OPENS	NONE	MAIN TURBINE TRIP IN 103 MINUTES
	AIR REMOVAL	AR-SPV-2C,D	LOSE CONDENSER VACUUM SLOWLY	MAIN TURBINE TRIP IN 103 MINUTES	
	BLEED STEAM	BS-V-39B VALVE BS-V-6B VALVE	BS-V-39B OPENS BS-V-6B OPENS	NONE	
	MAIN STEAM	MS-V-133B VALVE MS-V-137B VALVE	MS-V-133B OPENS MS-V-137B OPENS		
	HEATER DRAINS	BS-V-4B, 5B, 6B VALVE	BS-V-4B, 5B, 6B CLOSES	DECREASE IN FEEDWATER TEMPERATURE AND CONDENSER VACUUM	

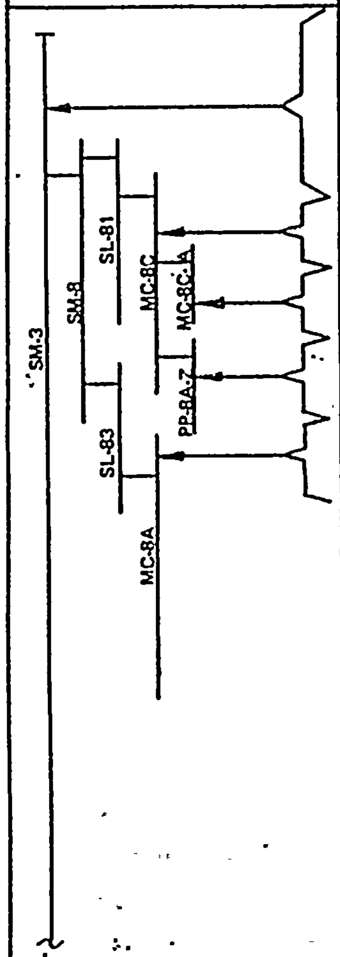
APPENDIX A

HANFORD - CONTROL SYSTEM FAILURE ANALYSIS

AC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
PP-2P-A	OFFGAS	OG-V-129B VALVE	OG-V-129B OPENS	NONE	NONE
	SEALING STEAM	SS-V-12B SS-V-18	SS-V-12B OPENS SS-V-18 OPENS	NONE.	
SL-21	MAIN STEAM	MS-V-125B	MS-V-125B OPEN	NONE	DECREASE IN FEEDWATER TEMPERATURE, DECREASE IN CONDENSER VACUUM
	BLEED STEAM	BS-V-58 VALVE BS-V-31 VALVE BS-V-52B VALVE	BS-V-58 OPENS BS-V-31 OPENS BS-V-52B OPENS	NONE	
	SEALING STEAM	SS-V-30 VALVE	SS-V-30 OPENS	NONE	
MC-2R MC-2R-A PP-2R-A-B	BLEED STEAM	BS-V-1B VALVE BS-V-8D BS-V-8E BS-V-1C BS-V-10D BS-V-12B BS-V-2C2 BS-V-8F BS-V-10E BS-V-2C1 BS-V-2C2 BS-V-3A2 BS-V-3C1 BS-V-10C BS-V-3C2 BS-V-4C BS-V-5B BS-V-2B1 BS-V-2B2 BS-V-3B1 BS-V-3B2 BS-V-4B BS-V-11F BS-V-12C BS-V-4C PLUS OTHER BS VALVES	BS-VALVES FAIL CLOSED OR OPEN PER VALVE TYPE	DECREASE IN FEEDWATER TEMPERATURE AND CONDENSER VACUUM	

APPENDIX A

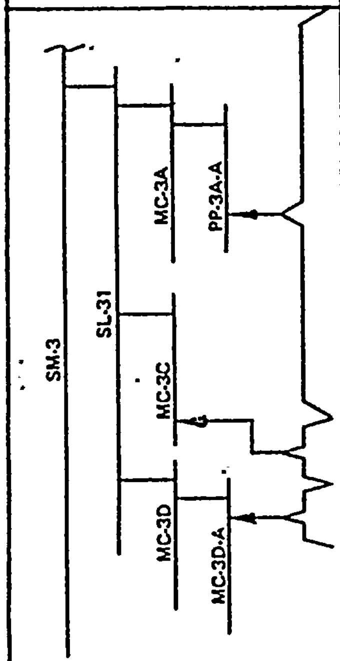
HANFORD CONTROL SYSTEM FAILURE ANALYSIS

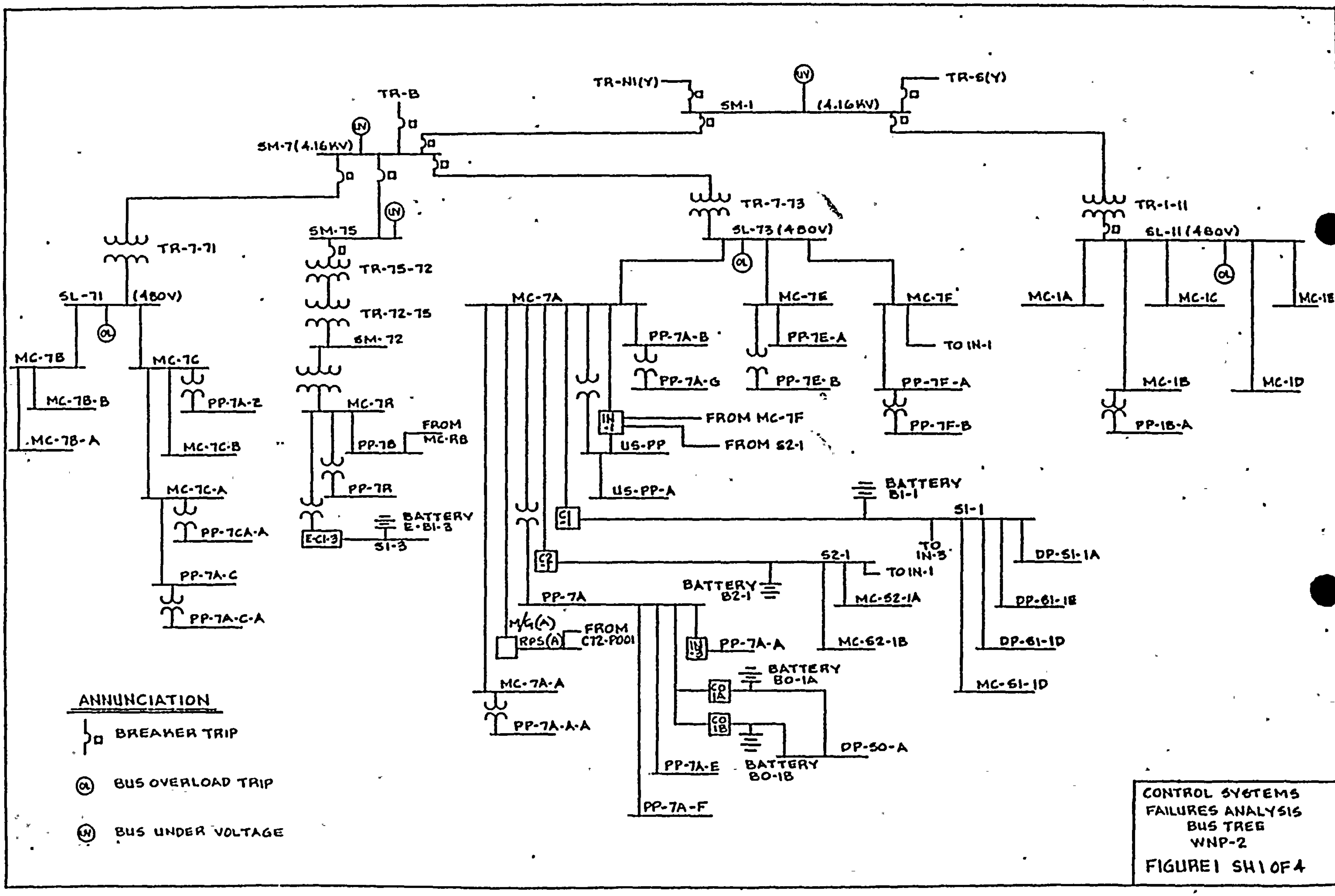


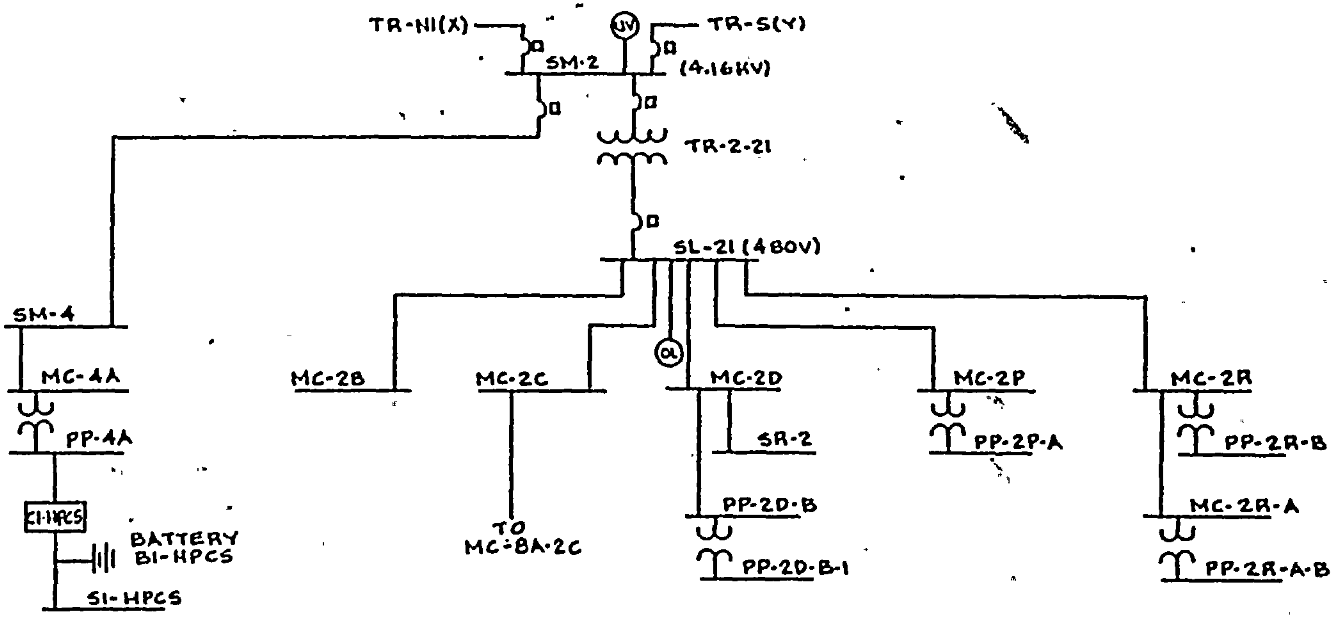
AC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
	REACTOR FEEDWATER	COND. PUMP 1C COND. BOOSTER PUMP 2C	PUMPS INOPERATIVE	DECREASE IN FLOW TO SUCTION OF RFP'S	SEE SECTION 4.0
	CIRC. WATER	CIRC. PUMP 1C		DECREASE IN CONDENSER COOLING FLOW	
	REACTOR RECIRC	LFMG SET S001B	LFMG SET S001B INOPERATIVE	NONE AT FULL POWER	
	REACTOR RECIRC	SUBLOOP HYD PWR UNIT D003B	IF OTHER LOOP NOT AVAILABLE FCV-60B LOCKS UP	NO LOAD FOLLOWING	NO LOAD FOLLOWING
	REACTOR RECIRC	SUBLOOP HYD PWR UNIT D003B	IF OTHER LOOP NOT AVAILABLE FCV-60B LOCKS UP	NO LOAD FOLLOWING	NO LOAD FOLLOWING
	REACTOR FEEDWATER	REACTOR FEEDWATER TURBINE GOVERNOR (RFPT-GOV-1B)	REACTOR FEEDWATER TURBINE "B" TRIP	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN BACK TO 68% POWER	RPV WATER LEVEL LOWER AND REACTOR RECIRC. RUN-BACK TO 68% POWER
	COMPRESSED AIR	INSTR. AIR COMPRESSOR 1B (COMP. 1A BUS ON SM-1, COMP. 1C BUS ON SM-2)	COMPRESSOR 1B INOPERATIVE	NONE, COMPRESSORS 1A & 1C MEET REQUIREMENTS	NONE

APPENDIX A

HANFORD CONTROL SYSTEM FAILURE ANALYSIS

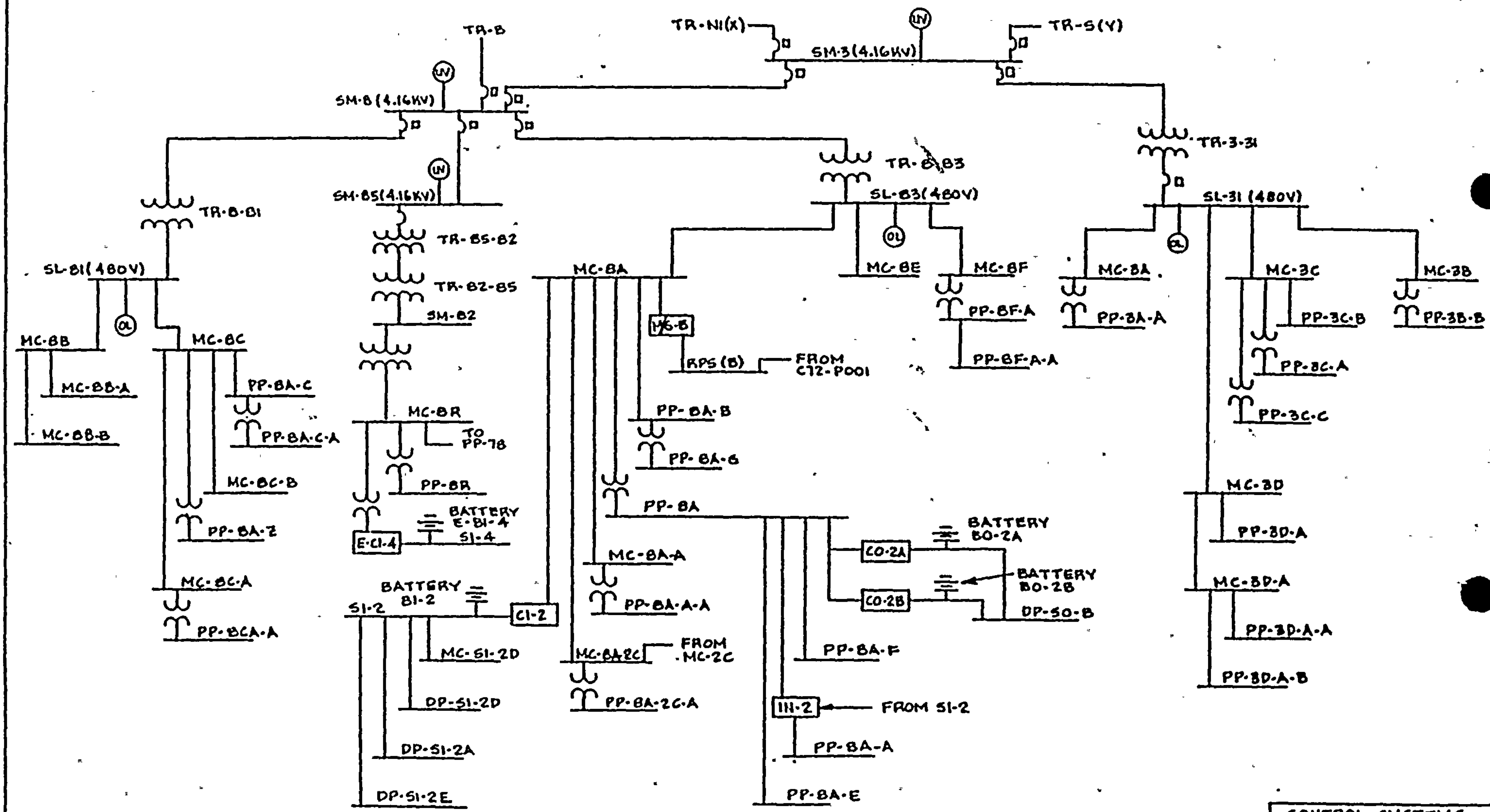
AC BUS	SYSTEM	COMPONENT DESCRIPTION	PRIMARY EFFECT	SECONDARY EFFECT	COMBINED EFFECTS
	BLEED STEAM	BS-V-8A,-8B BS-V-10A BS-V-10B BS-V-13A BS-DV-5A BS-DV-6A BS-DV-2A1 BS-DV-2A2 BS-DV-3A1 BS-DV-4A BS-V-27 BS-V-48 BS-V-52B BS-V-52A PLUS OTHER BS VALVES	BS-V-8A,B CLOSED IF FLOW LOST BS-V-10A CLOSED IF FLOW LOST BS-V-10B CLOSED BS-V-13A CLOSED BS-DV-5A OPEN BS-DV-6A OPEN BS-DV-2A1 OPEN BS-DV-2A2 OPEN BS-DV-3A1 OPEN BS-DV-4A OPEN BS-V-27 OPEN BS-V-48 OPEN BS-V-52B OPEN BS-V-52A OPEN	DECREASE FEEDWATER TEMPERATURE AND CONDENSER VACUUM	MAIN TURBINE TRIP ON HIGH OIL TEMPERATURE DECREASE IN FEEDWATER TEMPERATURE DECREASE IN CONDENSER VACUUM
	TURBINE SERVICE WATER	MAIN TURBINE OIL TEMPERATURE CONTROL VALVE	TSW-TCV-8 CLOSED	MAIN TURBINE TRIP ON HIGH OIL TEMPERATURE	
	OFFGAS	OG-V-125A OG-V-149A	DECREASE IN CONDENSER VACUUM	DECREASE IN CONDENSER VACUUM	DECREASE IN CONDENSER VACUUM
	REACTOR RECIRC.	SUBLOOP HYD PWR UNIT A D003A	IF OTHER SUBLOOP NOT AVAILABLE, FCV 60A LOCKS UP	NO LOAD FOLLOWING	NO LOAD FOLLOWING


TR-N1(Y)
SM-1
(4.16KV)
TR-S(Y)
TR-B
SM-7(4.16KV)
TR-7-73
SL-73(480V)
TR-1-11
SL-11(480V)
SM-75
TR-7-71
TR-75-72
TR-72-75
SM-72
SL-71
(480V)
MC-7A
MC-7E
MC-7F
MC-1A
MC-1C
MC-1E
MC-7B
MC-7C
PP-7A-B
PP-7E-A
TO IN-1
MC-7B-B
PP-7A-Z
MC-7R
PP-7A-G
PP-7E-B
PP-7F-A
MC-1B
MC-1D
PP-7F-B
PP-1B-A
MC-7B-A
MC-7C-B
PP-7B
FROM MC-RB
FROM MC-7F
US-PP
FROM S2-1
MC-7C-A
PP-7R
US-PP-A
BATTERY B1-1
PP-7CA-A
BATTERY E-B1-3
S1-1
E-C1-3
S1-3
TO IN-3
DP-S1-1A
S2-1
TO IN-1
PP-7A-C
BATTERY B2-1
PP-7A
MC-S2-1A
DP-S1-1B
PP-7A-C-A
M/G(A)
RPS(A)
FROM C72-P001
PP-7A-A
MC-S2-1B
DP-S1-1D
BATTERY B0-1A
MC-7A-A
MC-S1-1D
PP-7A-A-A
DP-S0-A
BATTERY B0-1B
PP-7A-E
PP-7A-F
ANNUNCIATION
BREAKER TRIP
BUS OVERLOAD TRIP
BUS UNDER VOLTAGE
CONTROL SYSTEMS
FAILURES ANALYSIS
BUS TREE
WNP-2
FIGURE1 SH1 OF 4



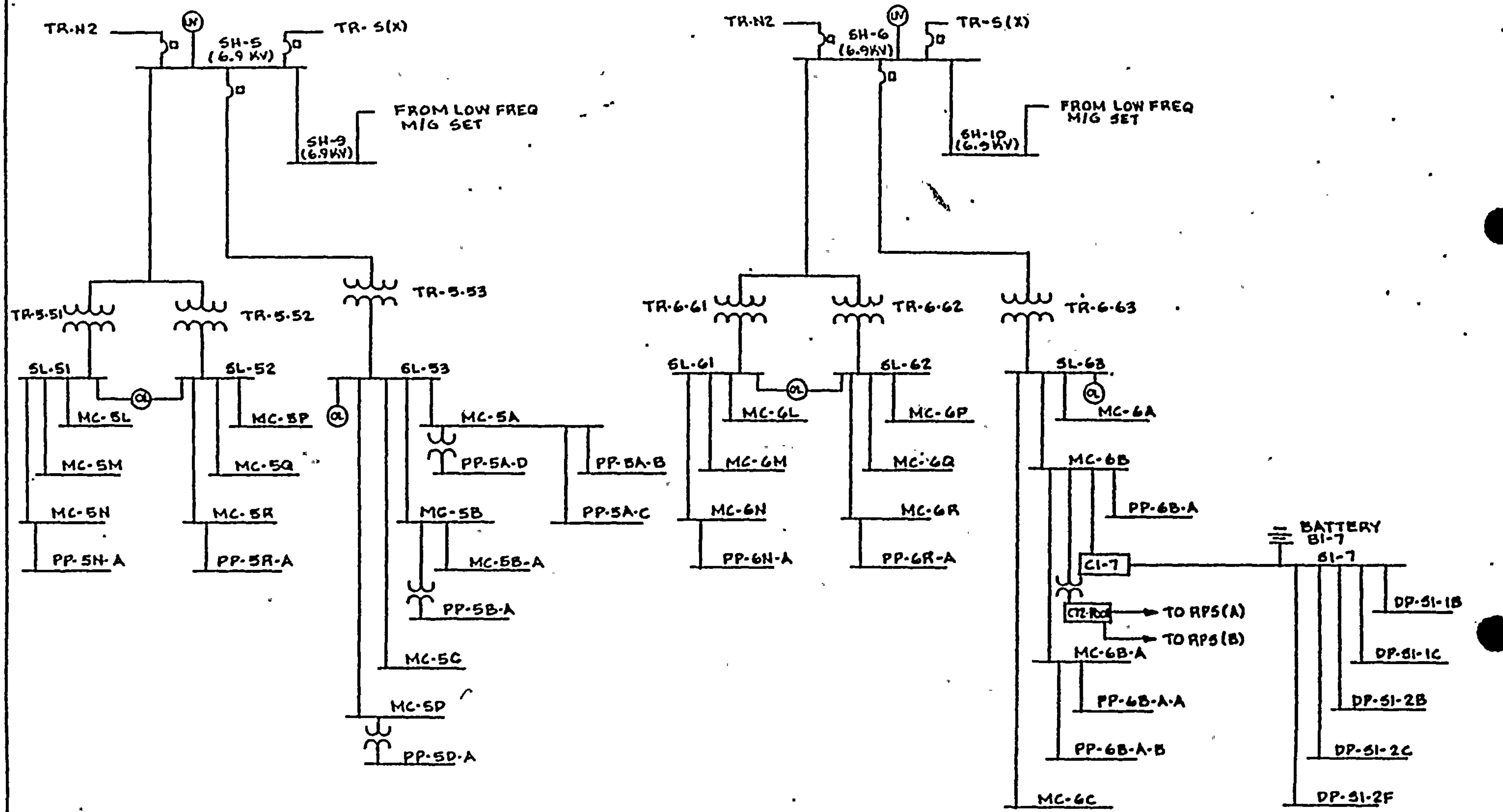
CONTROL SYSTEMS
FAILURES ANALYSIS
BUS TREE
WNP-2

FIGURE 1 SH2 OF 4



CONTROL SYSTEMS FAILURES ANALYSIS BUS TREE WNP-2

FIGURE 1 SH 3 OF 4



CONTROL SYSTEMS FAILURES ANALYSIS BUS TREE

WNP-2

FIGURE 1 SH 4 OF 4

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. Valves that fail as is and in a normal full open or close position are also in this category.
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.

POWER BUS (AE DESIGNATION)	SYSTEM (INCL. MPL#)	SYSTEM CONTROL AND INSTRUMENTATION LOADS ON BUS	GENERAL EFFECTS LIMITATIONS ON THE SYS.'S CAPABILITY TO PERFORM ITS PRINCIPAL FUNCTION DUE TO LOSS OF BUS	SPECIFIC EFFECTS OF BUS LOSS TO SYSTEM EFFECT ON SYSTEM SUBFUNCTIONS	EFFECTS ON OTHER SYSTEMS	** CODE	EFFECTS (WITHIN 10 MIN ON REACTOR WATER LEVEL, PRESSURE OR CPR	INPUTS REQUIRED FROM OTHER SYSTEMS AND EFFECT ON LOSS OF INPUT
Power Supply 399 in BD-GII (PP-8A-A) (CKT-33)	RFW (Reactor Feedwater B22)	Loop 04-A51 RFW-PT-1B PI-1B SRU-2 A0102	None	Indicator on BD-A & process computer indicate min. pressure	None N2	D	None	RFW-P-1B discharge pressure indication lost in control room.
		Loop 04-A53 RFW-FT-2B SQRT-2B SRU-8 FIC-2B E/P-2B	RFW-FCV-2B will go full open, by passing feedwater to main condenser feedpumps will increase to 115% flow to try to maintain feedwater flow to vessel	Low flow indication and signal to RFW-FIC-2B on BD-A causing min flow air to RFW-FCU-2B: (fully opening valve) M620-504.2	Reactor Vessel level will drop slowly, operator must take action to lower reactor power to prevent scram at Level 3	A	RPV level will drop	High condenser level annunciated in control room (Cond-LS-2H).
		Loop 04-T52 RFW-DPT-3B DPI-3B SRU-56	None	Indicator on BD-T indicates min. diff. pressure	None N2	D	None	ΔP across HP heater B indication lost.

APPENDIX C

REFERENCES:

FSAR 10.4, Table 10.4-2	M634 04-A51/2
M504, Rev. 36	M634 04-A53/3
M620/504-2, Rev. 1	M634 04-T52/0

**Code Classification for Effects on Reactor Parameters:
"A" - Immediate (<1 minute) and Direct
"B" - Immediate but Indirect
"C" - Effect is Delayed
"D" - No Effect on Reactor Parameters (<10 minutes)

hjr/C06294-44*