TITLE: EMERGENCY OPERATING PROCEDURE
REACTOR TRIP WITH SAFETY INJECTION

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the initial operating steps to be taken in the event of a reactor trip with safety injection signal. The safety injection signal may occur at some time after the reactor trip has taken place. If this is the case, the operator will cease using OP-5 (Reactor Trip Without Safety Injection) procedure and will use this procedure to control and analyze the plant condition.

SYMPTOMS[1]

The following symptoms are typical of those which may arise in a plant which is undergoing a loss of reactor coolant, loss of secondary coolant or steam generator tube rupture (one or more symptoms in each category may appear in any order)[2] :

LOSS OF REACTOR COOLANT

Lo Pressurizer Pressure

Lo Pressurizer Water Level

Hi Pressurizer Water Level

Letdown Isolation/Pressurizer Heater Cutout

Increased Charging Flow

Hi Containment Pressure

Hi Containment Temperature

Hi Containment Humidity

Hi Containment Radiation

Hi Containment Recirc. Sump Water Level

[1]The process variables referred to in this Instruction are typically monitored by more than one instrumentation channel. The redundant channels should be checked for consistency while performing the steps of this Instruction.

[2]The pressurizer water level indication should always be used in conjunction with other specified reactor coolant system indications to evaluate system response and to initiate manual operator actions.

LOSS OF SECONDARY COOLANT

Lo Pressurizer Pressure

Lo Pressurizer Water Level

Letdown Isolation/Pressurizer Heater Cutout

Lo-Lo Reactor Coolant Tavg

Hi Containment Pressure

Hi Containment Temperature

Hi Containment Recirc. Sump Level

Steam Flow/Feedwater Flow Mismatch

Lo Steam Line Pressure (one or all Steam Lines)

Lo Steam Generatcr Water Level

Hi Steam Flow (one or all Steam Lines)

Lo Feedwater Pump Discharge Pressure

STEAM GENERATOR TUBE RUPTURE

Hi Air Ejector Radiation

Lo Pressurizer Pressure

Lo Pressurizer Water Level

Increasing Charging Flow

Letdown Isolation/Pressurizer Heater Cutout

Steam Flow/Feed Flow Mismatch

Hi Steam Generator Blowdown Radiation

Increasing Stem Generator Water Level

AUTOMATIC ACTIONS

1. Reactor trip and turbine trip.

2. Safety injection initiated.

OBJECTIVES

1. To verify the reactor trip and safety injection.

NOTE: If the plant is in a condition which warrants a reactor trip and safety injection and an automatic reactor trip and safety injection has not yet occurred, it is the reactor operators responsibility to manually initiate the reactor trip and safety injection.

2. To verify all SI equipment is operating and performing the intended function.

3. To monitor plant parameters and diagnose the initiating SI signal.

4. To mitigate the consequences of a valid SI signal by providing direction once the initiating signal is identified.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Verify the following automatic actions. If required, use manual control to satisfy the action.	
a. Reactor trip (all rods on bottom, DRPI - Nuclear Instruments Decreasing.)	
b. Turbine trip (all four SV closed on EH panel.)	
c. Vital 4160 busses F, G and H transferred to startup power (breaker positions on VB-4.)	
d. Vital 4160 busses F, G and H voltage normal (120 volts indicated on 480 volt vital busses F, G and H, VB-4.)	
e. Diesel generators running and voltage normal (diesel RPM and generator voltmeter on VB-4.)	
f. Auxiliary building ventilation system in building and safeguards mode (mode light on ventilation section of VB-4).	
g. Control room ventilation system in mode 4 (mode light on VB-4).	
h. Both motor driven auxiliary feedwater pumps running and all 4 auxiliary feedwater LCV open. (Motor breaker position lights and LCV position indicators on VB-3).	

ACTION	COMMENTS
i. Check the safeguards postage stamp monitor lights on VB-1.	
IF 2 steam loops have Hi steam flow in alarm with any 2 loop Lo pressure in alarm	
OR 2 steam loops have Hi steam flow in alarm with any 2 loop Lo-Lo Tavg in alarm	
THEN: Verify all 4 main steam isolation valves, bypass valves and all 4 steam generator blowdown stop valves IC closed (valve position indication lights on VB-3).	
j. ALL ECCS pumps have started and automatic valve operations have occurred (all SI/FW ISOL/STM GEN LEVEL postage stamp monitor lights on VB-1 not in alarm).	j. Verify minimum of one ECCS train operating (SI, charging and RHR pump).
k. Containment Phase A and containment ventilation isolation (all containment Phase A postage stamp monitor lights on VB-1 not in alarm).	

SUBSEQUENT OPERATOR ACTIONS (PART A)

1. Verify the following pump flows.

   If the flows are not occurring, attempt to operate equipment manually or locally to establish the flows.

ACTION		COMMENTS	
a.	Minimum of one safety injection pump running with flow indicated when W.R. RCS pressure is <u>less</u> than the SI pump shutoff head (1465 psig). (VB-1)	a.	<u>NOTE</u>: If W.R. RCS Pressure is greater than the shutoff head of the SI pump, continue on with this procedure to monitor W.R. RCS pressure and verify SI pump flow if pressure drops below the shutoff head of the SI pump.
b.	Minimum of one charging pump running with flow via BIT indicated on FI 917. (VB-2)		
c.	Verify both motor driven AFP running and flow indicated, to all 4 steam generators (VB-3),		
<u>OR</u>	Start the steam driven AFP by opening FCV 95 and establish flow to all 4 steam generators (VB-3).		
	Maintain maximum AFW flow until the steam generator water levels are in the narrow range. When the SG water levels approach 33% NR, verify automatic steam generator level control.	c.	Automatic Steam Generator level control only applies to motor driven AFP's.
d.	Verify RCS heat removal by 1) Observing automatic dump to condenser or atmospheric steam dump via the 10% steam dump valves (VB-3). <u>AND</u> 2) RCS Tavg decreasing to no-load temperature 547°F (VB-2)	d.	Atmospheric steam dump will be blocked by an existing "Turbine Tripped" condition. If condenser steam dump has been blocked due to a control malfunction or loss of the "Condenser Available" condition, decay heat removal will be effected by automatic actuation of the steam generator 10% atmospheric steam dump valves, or if

ACTION

COMMENTS

these prove ineffective, the steam generator code safety valves. In this event, steam pressure will be maintained at the set pressure of the controlling valve(s) and reactor coolant average temperature will stabilize at approximately the saturation temperature for the steam pressure being maintained.

2. Monitor containment pressure (VB-1), if containment pressure reaches or exceeds 22 psig, verify the following actions. If required, use manual control to satisfy the action.

   a. Main Steam Isolation Valves and Bypass Valves closed. Phase B isolation postage stamp monitor lights not in alarm (VB-1).

   b. Steam Generator Blowdown Valves IC closed. Phase B isolation postage stamp monitor lights not in alarm (VB-1).

   c. Containment spray initiated (phase B isolation postage stamp monitor lights not in alarm VB-1).

   d. Phase B isolation (phase B isolation postage stamp monitor lights not in alarm VB-1).

   e. Manually trip all 4 RCP's.

   Comment e. CCW to the lube oil coolers will be lost on the phase B isolation.

3. Monitor the core exit thermocouple temperatures for indications of inadequate core cooling. If indications of inadequate core cooling exist, perform Appendix 3 of this procedure.

SUBSEQUENT OPERATOR ACTIONS (PART B)

ACCIDENT DIAGNOSTICS

1. Evaluate RCS pressure:

   a. If W.R. RCS pressure falls below or is below 1950 psig, close or verify closed the following valves.

      Pressurizer Spray Valve PCV 455A
      Pressurizer Spray Valve PCV 455B
      Auxiliary Spray Valves 8145 and 8148
      Pressurizer Power Relief Valve PCV 474
      Pressurizer Power Relief Valve PCV 456
      Pressurizer Power Relief Valve PCV 455C

      a. Verify closed by observing position indication lights and discharge pipe temperature indicators.

   b. If W.R. RCS pressure remains above 1950 psig and is stable or increasing, go to Step 1 of Part C Subsequent Operator Actions.

2. If W.R.RCS pressure continues to decay below 1220 psig or is below 1220 psig and stable.

   2. NOTE: The conditions for stopping RCP must be continuosly monitored throughout the transient.

   a. Again verify a minimum of one charging pump delivering flow and one SI pump delivering flow to the RCS

      a. SI flow rate will increase with decreasing RCS pressure.

   b. Then, STOP all four reactor coolant pumps. Maintain seal water flow to the RCP seals.

   c. Close the centrifugal charging pump recirculation valves, 8105 and 8106.

      c. NOTE: When the W.R. RCS pressure is restored above 2000 psig, reopen valves 8105 and 8106 to prevent pump damage.

   d. If component cooling water to the RCP's is isolated due to a containment phase B isolation, stop all RCP's within 5 minutes and maintain seal flow as above.

ACTION	COMMENTS
3. If condenser air ejector radiation monitor is reading abnormally high radiation AND containment pressure, containment area radiation monitor, and containment recirc. sump level exhibit NORMAL readings, discontinue this procedure and begin procedure OP-3A, Steam Generator Tube Rupture.	
4. If steam generator pressure is ABNORMALLY LOW in one steam generator as compared to the other steam generators, discontinue this procedure and begin procedure OP-2, Loss of Secondary Coolant.	4. This is indicative of a secondary break upstream of MSIV.
5. If containment pressure, containment area radiation monitor, or containment recirc. sump level exhibit ABNORMALLY HIGH or INCREASING levels, discontinue this procedure and begin procedure OP-1, Loss of Reactor Coolant.	5. NOTE: For very small coolant breaks inside the containment, the containment pressure and containment recirc. sump level may increase very slowly and possibly not recognizable by the operator immediately. Therefore, the operator should monitor these parameters throughout the transient.
6. If containment pressure, containment area radiation monitor and containment recirc. sump level remains stable in the pre-event range, discontinue this procedure and begin procedure OP-2, Loss of Secondary Coolant.	6. This is indicative of a secondary break downstream of MSIV.

SUBSEQUENT OPERATOR ACTIONS (PART C)

ACTION	COMMENTS
1. Assume the event is NON SPURIOUS safety injection until all of the following items are verified NORMAL.	1. This step is entered from Accident Diagnostics step 1 b.
a. Containment pressure.	
b. Containment temperature.	
c. Containment recirc. sump level.	
d. Containment area radiation monitor.	
e. Condenser air ejector radiation.	
f. Auxiliary Bldg. control board area radiation.	
g. Reciprocal charging pump room area radiation.	
h. Plant ventilation particulate monitor.	
i. Plant ventilation radio gas monitor.	
If the above items a. through i. cannot be verified NORMAL, return to step 1 of Accident Diagnostics.	
If the above symptoms a. through i. are normal, and when	
j. W.R. RCS pressure is greater 2000 psig.	
AND k. Pressurizer water level is greater than 22%.	
AND l. RCS indicated subcooling is greater than 35°F.	1. If the RCS subcooling meter is inoperable or is suspected to be incorrect, use wide range Thot in conjunction with the attached RCS saturation curve (graph) to determine RCS subcooling.

ACTION

AND m. Auxiliary feedwater flow to each steam generator is greater than 205 gpm or

One steam generator wide range water level instrument indicates a level greater than 75%.

THEN

2. Reset safety injection.

a. Reset containment isolation phase A, train A and train B.

b. Stop one charging pump at a time and evaluate RCS pressure. Maintain sufficient charging flow to supply adequate seal injection flow and to prevent RCS pressure from decaying. If RCS pressure drops below 1850 psig,

COMMENTS

2. CAUTION: 1) Automatic reinitiation of safety injection will not occur after this step since the reactor trip breakers are open. If the operator has indication that an SI is required after this step, he must initiate it manually.

CAUTION: 2) If loss of off-site power occurs after resetting safety injection, it will be necessary to load the safeguards equipment onto the vital busses manually. If safety injection is reinitiated manually after the loss of off-site power the vital busses will automatically sequentially load the safeguard equipment.

If manual loading or automatic loading is performed, verify the equipment given in Appendix A is loaded onto the vital busses.

ACTION | COMMENTS

reinitiate SI and return to Step 1 of the diagnostics.

c. Stop both RHR and SI pumps.

d. Verify AC turbine bearing oil pump and Hi pressure seal oil backup pump running after oil pressure decays to 11 psig and turbine bearing lift pump starts at 600 RPM turbine speed.

e. Establish normal charging.

1) Open instrument air valves FCV-584 and 682.

2) Check open or open normal charging valve 8146.

3) Check close or close charging to auxiliary spray valves 8145 and 8148 and alternate charging valve 8147.

4) Open charging line isolation valves MO 8107 and 8108.

5) Adjust HCV-142 and FCV-128 or reciprocal charging pump speed to achieve RCP seal flow and charging flow as required to maintain pressurizer level greater than 22%.

6) Open RCP seal return valves MO 8100 and 8112. Check RCP seal return flow normal.

7) Close the BIT inlet and outlet valves 8803A and B, 8801A and B.

ACTION

3. Verify the following:

   a. With normal RCS charging, the pressurizer water level remains above 10%.

   b. W.R. RCS pressure remains above 1850 psig.

   c. RCS indicated subcooling greater than 35°F.

   If item a., b., or c. above cannot be verified MANUALLY, REINITIATE SAFETY INJECTION and return to diagnostic Step 1, subsequent action Part "B", of this procedure. If the ECCS pumps are restarted, add 15°F to item c. above.

4. Verify auxiliary feedwater flow and steam generator water levels approaching NO LOAD level (33% narrow range).

5. Establish normal letdown.

   a. Check open or open letdown valves LCV-459 and 460.

   b. Open letdown isolation valve 8152.

   c. Open one 75 gpm letdown orifice valve.

   d. Verify PCV-135 opening by observing letdown flow.

6. Establish VCT makeup and transfer charging pumps suction to VCT.

   a. Adjust VCT makeup blend to the existing boron concentration.

   b. Open VCT outlet valves LCV-112B and C.

   c. Close RWST to charging pump suction valves 8805A and B.

   d. Verify divert valve LCV-112A in AUTO.

   e. Verify charging flow normal.

COMMENTS

c. CAUTION: Stopping and starting of the ECCS pumps can cause pump motor overheating or reduced motor life.

ACTION	COMMENT
7. If RCP's are not running, establish conditions for starting RCP's and start at least one RCP.	7. Start RCP 1 or 2 if possible to provide Pressurizer Spray capability.
8. With pressurizer level controlled in manual, verify pressurizer pressure control in AUTO and pressurizer sprays and heaters controlling pressure.	
9. Stop all 3 diesel generators, place diesel generator control switches in AUTO.	
10. Insure the main and feedpump turbines on turning gear once 0 RPM speed is reached.	
11. <u>IF</u> after securing safety injection and transferring the plant to normal pressurizer pressure and level control, the reactor coolant pressure does not drop below the low pressurizer pressure setpoint for safety injection actuation <u>AND</u> the pressurizer water level remains above 10%, <u>AND</u> the reactor coolant indicated subcooling is greater than 35°F, <u>then consider the event</u> a spurious safety injection. Continue to monitor these parameters closely; if any parameter fails to remain above the limit, manually reinitiate Safety Injection and return to Step 1 of the Diagnostic section.	11. <u>CAUTION</u>: Do not reset the reactor trip breakers until authorized by the Plant Superintendent.

<u>SPURIOUS SI SIGNAL RECOVERY</u>

1. Proceed to a normal Hot Standby condition as follows:	
a. If steam line isolation has occurred:	
1) Close or check closed all 35 and 40% steam dump valves.	
2) Prepare or verify main condenser available for service.	
3) Equalize or attempt to equalize and open all 4 MSIV's. Monitor steam generator pressure closely during this operation. Immediately close all MSIV's if steam pressure rapidly drops during this operation.	3) A steam line break downstream of MSIV's will prevent equalizing.
4) Establish steam dump to condenser using steam pressure mode set at 1005 psig.	

ACTION | COMMENTS

5) If steam dump to the condenser is established, verify all atmospheric steam dump valves closed.

b. Open the main generator motor operated disconnect switch and reenergize the unit auxiliary transformers by back-feeding from the 500 KV yard. Transfer all station auxiliary busses (12 and 4 KV busses) to the unit auxiliary transformers.

c. Return the auxiliary building ventilation system to normal by resetting the "S" signal on both POV cabinets and selecting building only mode on VB-3.

d. Reset both Units 1 and 2 control room ventilation systems on Unit 2 radiation control board and verify both Units 1 and 2 ventilation systems return to the normal mode of operation.

e. When directed by the SFM, shutdown the following:

1) One auxiliary saltwater pump.

2) Steam driven auxiliary feedwater pump after steam generator levels are greater than 33% and motor driven pumps are controlling level.

3) Close CFCU maxi flow valves then shutdown 1 CCW pump.

4) Remove 2 CFCU's from service and place the running CFCU's on fast speed.

f. Verify BA transfer pump running and open the BIT recirc. valves to begin increasing BIT concentration. Call Chemical and Radiation Department to begin sampling the BIT and BAT.

g. Reset containment ventilation isolation trains A and train B.

ACTION

COMMENTS

1) Open containment rad. Gas monitor valves FCV-681, 678 and 679.

2) Verify normal readings on containment air particulate and radio gas monitors on RMS board.

h. Reset radwaste isolation valves reset switches.

i. Open fire water valve FCV-633 and primary water to containment valve 8029.

j. Open valve 8045 N_2 to PRT.

k. If the incore chiller has been in service prior to the SI, open incore chiller valves FCV 655, 657, 654 and 656.

l. If the gross failed fuel detector has been in service prior to the SI, open Hot Leg sample valves 9356A and B.

1) Verify flow returns on the GFFD flowmeter.

2) Verify the GFFD countrate returns on scale and stabilizes at a value below the post SI countrate on the recorder.

m. Open the pressurizer steam space sample valves 9354A and B

n. Open pressurizer relief tank gas analyzer valve 8034A.

2. Maintain Hot Standby conditions until authorized to proceed with a normal startup or inform the plant superintendent that the unit is proceeding to Cold Shutdown.

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

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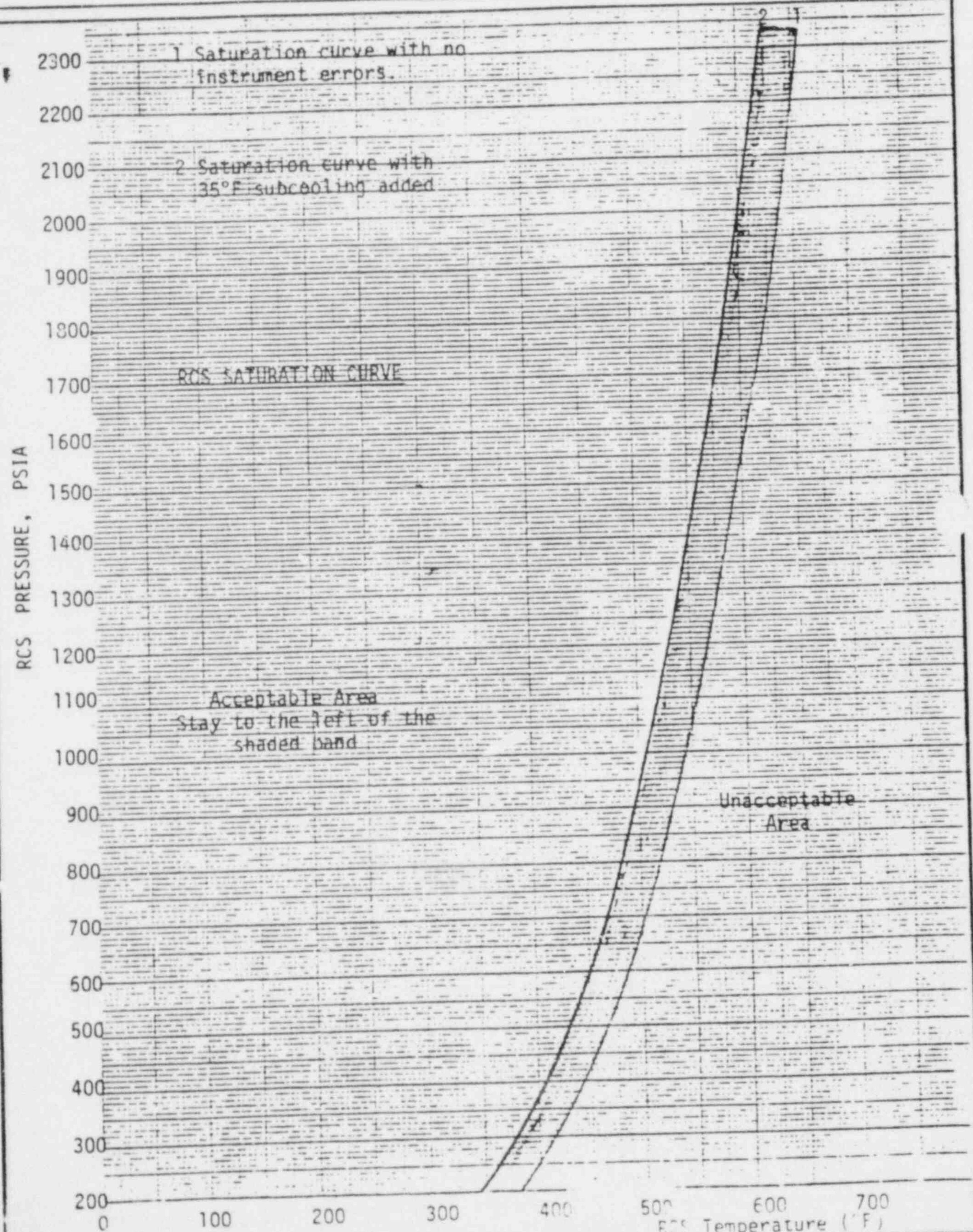
TITLE: REACTOR TRIP WITH SAFETY INJECTION



APPENDIX A

BLACKOUT WITH SAFETY INJECTION EMERGENCY LOADING OF VITAL BUSSES

1. If the vital busses lose voltage prior to resetting the safety injection signal, the vital busses will automatically load the vital equipment given below. Verify the equipment has been loaded by observing breaker lights on the control board.

2. If the vital busses lose voltage after the safety injection signal has been reset, load or verify loaded the equipment given below onto the vital busses manually. Allow approximately 4 seconds between loading of each piece of equipment onto a given vital bus. Load or verify that the CFCU are running in Low Speed.

VITAL BUS F	VITAL BUS G	VITAL BUS H
D/G 1-3	D/G 1-2	D/G 1-1
MCC 1-F	MCC 1-G	MCC 1-H
CC Pp 1-1	CC Pp 1-2	SI Pp 1-2
SI Pp 1-1		
CFCU 1-2	RHR Pp 1-1	RHR Pp 1-2
CFCU 1-1	CFCU 1-3	CFCU 1-4
CCW Pp 1-1	CFCU 1-5	CCW Pp 1-3
ASW Pp 1-1	CCW Pp 1-2	AFW Pp 1-2
AFW Pp 1-1	ASW Pp 1-2	

3. Load the containment spray Pumps only if they were running prior to the blackout.

VITAL BUS G	VITAL BUS H
Cont Spray Pp 1-1	Cont Spray Pp 1-2

APPENDIX B

DETERMINATION OF ADEQUATE CORE COOLING

This appendix provides the guidance to determine adequate core cooling if inadequate core cooling is suspected. Further, the instructions for regaining adequate core cooling is presented.

ACTION	COMMENTS

1. Monitor the core exit thermocouple temperatures.
   a. If the P-250 is available go to step 2.
   b. If the P-250 is not available go to step 3.

2. If 5 or more P-250 thermocouple readings exceed 1200°F, notify the Shift Foreman that inadequate core cooling exists and go to step 5.

   If there are not 5 or more that exceed 1200°F, discontinue this appendix but continue to monitor the thermocouple readings.

3. At the thermocouple incore board in the control room, Monitor 10 core-centered thermocouples. If any 3 of the 10 thermocouples exceed 700°F (pegged hi) proceed to step 4.

   If there are not 3 thermocouples that exceed 700°F, discontinue this appendix but continue to monitor the thermocouple readings.

4. If SI flow to the RCS and AFW flow to the steam generators cannot be verified, notify the Shift Foreman that inadequate core cooling may exist and go to step 5. If both SI flow to the RCS and AFW flow to the steam generators can be verified, discontinue this Appendix but continue to monitor the thermocouple readings.

5. The Shift Foreman will verify if inadequate core cooling exists using the appropriate steps above. If inadequate core cooling exists the Shift Foreman will direct operations as follows:

APPENDIX B (Cont.)

ACTION	COMMENTS
a. Declare a General Emergency Implement the instructions given in Emergency Procedure G-1 regarding on and off-site protective actions.	
b. Attempt to establish SI flow to the RCS and AFW flow to the steam Generators.	
c. Continue monitoring core outlet temperature to determine the effectiveness of the remaining actions.	
d. DEPRESSURIZE THE RCS by method 1 or 2 below.	
1) Dump steam to the condenser or atmosphere if the steam generator levels are in the narrow range and AFW flow is evidenc.	1) THIS IS THE PREFERRED METHOD.
2) Verify the SIS or charging pumps are running and available to deliver water to the RCS. THEN Open the pressurizer PORV's.	2) Opening the PORV's will provide a drop in RCS pressure sufficient to allow the SI flow required to cool the core. This method is to be used only if 1) (above) is ineffective.
e. If no means of depressurization are available, or if the depressurization did not result in decreasing core exit thermocouple temperatures, THEN START one RCP if possible. If the RCP fails after starting, replace the lost RCP with any remaining RCP.	e. Attempt to establish CCW and seal water flow to the pump; however, if CCW and/or seal water flow cannot be established, proceed to start a RCP. The pump must be started to move coolant thru the core.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been implemented, and upon direction from the Shift Foreman, proceed as follows:

   a. Designate this event a Nofification of Unusual Event if ECCS flow is indicated. Notify plant staff and response organizations required for this classification by implementing Emergency Procedure G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation".

   b. Reclassify this event according to the Appendix Z instructions in OP-3A "Steam Generator Tube Rupture" or OP-1 "Loss of Coolant Accident", if the accident diagnostics require implementation of either these procedures.

   c. In the event inadequate core cooling is verified per Appendix B reclassify this event as a General Emergency. Notify plant staff and response organizations required by EP G-2 and G-3 in accordance with EP G-1.

TITLE: EMERGENCY OPERATING PROCEDURE
LOSS OF COOLANT ACCIDENT

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the operating steps to be taken in the event of a Loss of Coolant Accident. It is assumed that reactor trip and safety injection actuations have occured. The operator should have already performed the immediate operator actions and diagnostics of Emergency Operation Procedure No. OP-0.

SYMPTOMS

(See OP-0 Symptoms/Diagnostics)

AUTOMATIC ACTIONS

(See OP-0)

OBJECTIVES

1. Verify and establish short term core cooling to prevent or minimize damage to the fuel cladding and release of radioactivity.

2. Maintain long term shutdown and cooling of reactor by recurculation of spilled reactor coolant, injected water and containment spray drainage.

3. In the case of the small break LOCA, to reduce plant conditions to the cold shutdown condition for repairs.

IMMEDIATE OPERATION ACTIONS

1. Perform the immediate operator actions in the reactor trip with Safety Injection Emergency Procedure OP-0.

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. Initiate the site Emergency alarm.	
2. Monitor the RWST level	
a. If the level decreases slowly, proceed to step 3.	a. A small break LOCA is indicated.

ACTION	COMMENTS
b. If the level decreases rapidly such that the low level alarm is eminent, verify ECCS flow from at least one charging and one SI pump, and after W.R. RCS pressure decreases to <1220 psig, stop all 4 reactor coolant pumps and close the Centrifugal charging pump recirc. valves (8105 & 8106). Proceed to Appendix A, LOCA Injection/ Recirculation changeover procedure.	b. A large break LOCA is indicated. The RCP seal water flow should be maintained when the pumps are tripped. The recirc. valves (8105 & 8106) shall be reopened if pressure is restored to >2000 psig to protect the pumps from damage.
3. Monitor the RWST level closely <u>throughout</u> this procedure. If <u>RWST</u> level is decreasing, such that the low level RHR pump trip setpoint 33% appears imminent, go directly to Appendix A, LOCA Injection/ Recirculation Changeover Procedure, attached. When Appendix A is completed, return to this procedure and continue.	3. In the worst case, with RCS near 0 psig and minimum Tech. Spec. level in RWST with all ECCS equipment discharging including containment spray, the Operator has 14 minutes from time of break to RWST lo level.
4. Verify containment sump level increasing. If no increase is evident as the RWST level decreases, return to OP-0 diagnostics, step 3, 4, 5 and 6, to reevaluate the accident.	4. <u>CAUTION</u>: In the case of the samll break, the sump may rise slowly as the RWST level decreases slowly. Monitor the containment sump along with the RWST. If the containment sump fails to increase with decreasing RWST, the accident may have been misdiagnosed.
5. Verify steam generator levels being maintained and flow approximately equal to all steam generators. If any steam generator level increases in an unexpected manner, go to OP3A (Steam Generator Tube Rupture).	5. Steam generator tube ruptures may be indicated.
6. Monitor the condensate storage tank and upon reaching approximately 10% level, perform a. or b. below	6. If the lo lo level alarm occurs on the CST, the operator has approximately 25 minutes to perform item a. or b.
a. Verify a level in the raw water storage reservoir; then open FCV 436 and 437 (reservoir supply to AFW pumps). Allow the AFW pumps to run during the transfer.	

ACTION	COMMENTS
Monitor the AFW flow closely. If flow is lost, trip all 3 AFW pumps until the transfer is complete, then restart the pumps.	
b. If the raw water storage reservoir is not available, go to Appendix C (AFW Pump Suction Supply from Fire Water Tank Procedure). Allow the AFW pumps to run during the transfer. Monitor the AFW flow closely. If flow is lost, trip all three AFW pumps until the transfer is complete, then restart the pumps.	
7. Monitor the core exit thermocouple temperature for indications of inadequate core cooling. If indications of inadequate core cooling exit, perform Appendix F of this procedure.	
8. Verify the following:	
a. PORV PCV 474 closed PORV PCV 456 closed PORV PCV 455C closed	a. Verify by position indication and discharge pipe temperature indicators.
b. PORV Backup Valve 8000A open PORV Backup Valve 8000B open PORV Backup Valve 8000C open	b. Verify, by position indication lights, valves open and power available.
c. If any PORV opens during this procedure, verify closure. If the valve fails to close, close the backup valve.	
d. Monitor W.R. RCS pressure. If RCS pressure remains <2000 psig and stable or decreasing, go to step 15.	

ACTION	COMMENTS
9. If W.R. RCS pressure begins to increase after step 8. above, maintain full SI flow until the SI Termination Criteria below is met. Continue to monitor throughout this procedure for the below SI Termination Criteria. a. W.R. RCS pressure >2000 psig and increasing b. AND, PZR level >50% c. AND, RCS indicated subcooling >35°F	9. If the centrifugal pump recirc. valves (8105 & 8106) were closed due to low RCS pressure (1220 psig), reopen the valves when the RCS pressure increases above 2000 psig. NOTE: Pressure rising indicates the problem could have been a stuck open PORV. It may take a minute or two after step 8 for the pressure change. c. If at any time the subcooling Margin Monitor becomes inoperable or is suspect, use the saturation curve to determine subcooling.
d. AND, All 4 steam generator NR water levels are greater than 33% OR Flow to all lo level steam generator is greater than 205 gpm per lo level steam generator.	
10. If the SI termination criteria in step 9 CANNOT be met, go to step 15.	
11. If the above SI termination criteria IS met, RESET SAFETY INJECTION and proceed as below:	11. CAUTION 1: Automatic reinitiation of safety injection will not occur after this step since the reactor trip breakers are open. If the operator has indication that an SI is required after this step, he must initiate it manually. CAUTION 2: If loss of offsite power occurs after resetting safety injection, it will be necessary to load the safeguards equipment onto the vital buses manually. If safety injection is reinitiated manually after the loss of offsite power, the vital buses will automatically sequentially load the safeguard equipment.

ACTION	COMMENTS
	If off-site power is lost after the SI signal is reset, go to Appendix G (Blackout with SI Emergency Loading of Vital Buses).
a. Reset containment isolation phase A.	
b. Stop both RHR and SI pumps.	
c. Stop one charging pump at a time and evaluate RCS pressure. Maintain sufficient charging flow to maintain RCS pressure and seal water injection flow. If W.R. RCS pressure drops below 2000 psig or PZR level drops below 50% or RCS subcooling drops below 35°F, reinstate sufficient charging flow to maintain RCS pressure. If W.R. RCS pressure drops below 1850 psig return to step 1 of the Accident Diagnostics in OP-0. If charging pumps are restarted after this step, add 15°F of subcooling to the SI termination criteria every time the pumps are restarted.	
d. Verify AC turbine bearing oil backup pump and hi pressure seal oil pump running after oil pressure decays to 11 psig and turbine bearing lift pump starts at 600 RPM turbine speed.	d. Verify the unit goes on turning gear at 0 RPM.
e. Establish charging flow to maintain RCS pressure.	e. Charging at a rate greater than normal might be required in order to maintain PZR pressure and level.
1) Open instrument air valves FCV-584 and 682.	
2) Check open or open normal charging valve 8146.	
3) Close or check closed charging to auxiliary spray valves 8145 and 8148 and alt. charging valve 8147.	
4) Open charging line isolation valves 8107 and 8108.	

ACTION	COMMENTS
5) Adjust HCV-142 and FCV-128 or reciprocal charging pump speed to achieve RCP seal flow and charging flow as required to maintain pressurizer level greater than 22%.	5) Automatic PZR level control may not be possible at this time due to the TAVG signal inaccuracies (if RCP's are tripped), or TAVG being out of the normal control band.
6) Open RCP seal return valves, MO 8100 and 8112. Check seal return flow normal.	
7) Close the BIT inlet and outlet valves 8803A and B and 8801A and B.	
f. Continue to monitor the Primary System. If W.R. RCS Press drops below 1850 psig or PZR water level drops below 22% or RCS indicated subcooling drops below 35°F, reinitiate SI manually and return to OP-0 diagnostics unless this has already been performed. If SI is reinitiated add 15°F to the SI TERMINATION criteria prior to the second termination of the pumps.	
g. Establish normal letdown	
1) Verify containment area radiation monitor reading approximately post-accident level.	1) Containment area hi radiation may mean fuel damage. If cont. hi radiation exists, do not establish letdown outside containment. Establish letdown to the PRT if letdown is required.
2) Open letdown valves LCV-459 and 460.	
3) Open letdown isolation valve 8152.	
4) Open one 75 gpm letdown orifice valve.	

ACTION	COMMENTS
5) Verify PCV-135 opening by observing letdown flow.	5) Monitor PZR level and W.R. RCS Press to ensure they remain >22% and 1850 psig during this step.
h. Establish VCT makeup and transfer charging pumps suction to VCT.	
1) Adjust VCT makeup blend to the value indicated on the boron measurement system.	1) This assures the primary system will not be diluted during the recovery actions.
2) Open VCT outlet valves LCV-112B and C after VCT level normal.	
3) Close RWST to charging pump suction valves 8805A and B.	
4) Verify divert valve LCV-112A in AUTO.	
i. Verify at least one RCP running or verify a bubble in the PZR and establish conditions for starting at least one RCP. If conditions for starting a RCP exists, start <u>at least</u> one RCP. Attempt to start <u>RCP 1 or 2</u> so that pressurizer spray will be available.	i. The criteria given in OP-0 for stopping the RCP's on low pressure <u>still applies</u>: Check SI pump flow and stop RCP's at W.R. RCS pressure 1220 psig. Also, if pressure drops below 1220 psig close the centrifugal charging pump recirc. valve (8105 & 8106). The recirc. valve shall be reopened if pressure is restored above 2000 psig.
j. Stop all 3 diesel generators if off-site power is available and place diesel generator control switches in AUTO.	
k. With pressurizer level greater than 22%, verify pressurizer heaters controlling pressure.	
l. Continue to monitor the condensate storage tank level and transfer the AFW pumps to an alternate source if low level is approached.	

ACTION	COMMENTS
12. Verify using the RCS subcooling monitor that 50°F subcooling is present.	
13. If 50°F subcooling is present and can be maintained, proceed witha normal cooldown using Operating Procedure L-5, if required, to perform repairs. If repairs are not required, maintain hot standby conditions until authorized to proceed by the Plant Superintendent.	13. The criteria tripping reactor coolant pumps on low pressure will not apply if a cooldown using Operating Procedure L-5 is performed.
If 50°F subcooling <u>is not</u> indicated, attempt to establish 50°F subcooling by steam dump from the steam generators to the condenser or atmosphere.	
a. If steam dump is necessary, reduce the steam generator pressure to maintain a RCS cooldown rate of less than 50°F/HR, consistent with the CVCS makeup capability.	a. If steam dump at the reduced pressure setting is used to maintain subcooling, proceed to cooldown the plant using Operating Procedure L-5 (Plant Cooldown from Minimum Load to Cold Shutdown).
b. If 50°F subcooling can be achieved with steam dump, proceed with the cooldown using L-5. If 50°F subcooling cannot be maintained go to step 15.	
14. If 50°F RCS indicated subcooling cannot be established, REINITIATE SAFETY INJECTION and return to OP-0 Diagnostics, step 1.	
15. If closing the PORV's in step 8 failed to recover the RCS pressure, maintain MAX SI flow and proceed to step 16.	15. If this is the case, then we probably have an unisolatable break in the primary system.
If SI termination criteria in step 9 could not be met, maintain MAX SI flow and proceed to step 16.	
If 50°F subcooling cannot be maintained in step 13 above, REINITIATE SI and proceed to step 16.	

ACTION	COMMENTS
16. If any SI equipment has failed, attempt to operate the equipment locally or from the control room. Effect repairs if necessary.	
17. If wide range RCS pressure is above 165 psig, stable or increasing, reset safety injection and stop both RHR pumps.	17. Restart the RHR pumps if RCS pressure decays to 165 psig after this step.
18. If the RCP's are operating, stop all 4 RCP's <u>after</u>:	18. If CCW is lost to the RCP's, stop all pumps within 5 minutes and maintain seal water flow. Monitor the conditions for stopping RCP's throughout this procedure. Conditions for stopping RCP's should continually be monitored during this procedure.
a. Verifying at least one charging and SI pump are delivering flow and b. W.R. RCS pressure decreases to less than 1220 psig.	
19. If W.R. RCS pressure decreases below the steam generator's pressure, verify RWST level decreasing rapidly. Maintain maximum SI flow until the RWST Lo Level RHR Pump Trip occurs (trip the RHR pumps at 33% if the automatic pump trip fails to occur), then perform Appendix A.	19. A large break LOCA is indicated.
20. If W.R. RCS pressure is decreasing or stable and is above 1065 psig, continue with the steps below.	20. A small break LOCA is indicated. The instructions given in step 20 will aid in the cooldown and depressurization.
a. Reduce steam generator pressure to 865 psig using steam dump by methods b or c below.	a. Reducing steam pressure will lower the RCS pressure and temperature.

ACTION	COMMENTS
b. If the condenser is available, open the main steam isolation valves and dump steam to the condenser using the steam header pressure mode.	b. Continue to monitor RWST level. As RCS pressure decreases SI flow will increase and RWST level decrease will accelerate. If pressurizer level has been lost, monitor PZR level indication for its return.
c. If the condenser is not available, use the 10% atmospheric dumps.	
d. Maintain a RCS cooldown no greater than 50°F/HR.	
21. Monitor RWST level closely. If the RWST Lo Level Alarm is imminent, perform step 24.e. If the alarm is not imminent, continue with step 22.	21. The cooldown should increase SI flow and decrease RWST level.
22. If containment Hi-Hi pressure has initiated containment spray:	
a. Verify the additive tank level is decreasing and that the entire tank is discharged.	
23. Periodically check the auxiliary building radiation monitors for ECCS leakage if Appendix A has been performed. If leakage is found, attempt to isolate the leakage; however, do not interrupt ECCS flow at any time.	
24. If Appendix A, LOCA Injection/Recirculation Changeover Procedure, has <u>not</u> yet been performed, continue to monitor the RWST level and <u>prepare</u> for Appendix A in the following manner.	
a. At the 480 volt vital load centers F and H, close the <u>breakers</u> for the following valves: 8980 RHR pump supply from RWST 52-1F-31. 8976 SI pump supply from RWST 52-1H-20.	a. This will allow operation of valves during Appendix A. <u>DO NOT</u> make any valve operation at this time.

ACTION	COMMENTS
b. Reset the SI, Phase B isolation and containment spray isolation signals.	b. Reset only. Do not start or stop any SI equipment.
c. Close (CUT IN) the series contactors for the following valves using the toggle switches in the control room. (VB-2) 8809A and B, RHR pumps, Injection to cold legs. 8974A and B, S.I., Pump Recirc. 8982A and B, RHR pumps suction from containment sump.	c. This will allow operation of valves during Appendix A. <u>DO NOT</u> make any valve operations at this time.
d. If the RCS pressure is greater than 1500 psig (shutoff head of the SI pumps), shutdown these pumps prior to transferring to cold leg recirculation.	d. During cold leg recirculation, no recirc. path is available to these pumps.
e. Maintain maximum SI flow <u>until</u> the RWST low level RHR pump trip occurs. (Trip the RHR pumps at 33% RWST level if the automatic trip fails, <u>then</u> <u>perform</u> Appendix A.)	

DO NOT PROCEED BEYOND THIS STEP UNTIL APPENDIX A HAS BEEN PERFORMED.

ACTION	COMMENTS
25. If containment spray has been initiated, prepare to provide RHR spray as follows in steps 26 through 31.	25. Containment spray should continue for 2 hours using RHR for iodine removal. If an RHR Train has failed, terminate containment spray. DO NOT use the operating RHR Train for containment spray.
26. Check that the entire content of the spray additive tank has been injected.	
27. Stop both containment spray pumps on receiving the RWST level lo lo alarm. Verify tank empty on RWST indicators.	
28. Close the No. 1 RHR to Cold Leg Injection Valve (8809 A).	

ACTION

29. Open the RHR Pump 1-1 to Spray Ring Valve (9003 A).

30. Close the Containment Spray Pump Discharge Valves (9001A and B).

31. Containment spray may be terminated after the 2 hour period when the containment pressure has been reduced to approximately 0 psig.

32. To terminate spray, proceed as follows:

    a. Close the RHR Pump 1-1 to Spr ' Ring Valve (9003 A).

    b. Open the No. 1 RHR to Cold Leg Injection Valve (8809 A).

    c. Regulate No. 1 RHR pump flow using its hand control valve (HCV-638) to maintain pump motor current less than 57.5 amps. (VB-1)

33. The systems are now lined up for the cold leg recirculation phase. Continue to operate in this manner for 19-1/2 hours.

    During this period, make the following motor operated valves available by closing their 480V breakers at the vital load centers.

    8802A SI pump No. 1 discharge to hot legs 52-1F-48.

    8835 SI pump common discharge to cold legs 52-1G-24.

    8703 RHR pumps common discharge to hot legs 52-1G-56.

    8802B SI pump No. 2 discharge to hot legs 52-1H-26.

COMMENTS

33. DO NOT make any valve changes after making the valves available.

ACTION | COMMENTS

34. During the cold leg recirculation phase, the Shift Foreman will form the plant operators into critical plant systems and equipment assesment teams. The systems or equipment to receive the greatest attention will be determined by the Shift Foreman and will be based on knowledge of past system/equipment reliability and service. This assesment function may be performed inside or outside the control room as conditions warrant. The critical parameters to be assessed will be determined by the Shift Foreman. The Shift Foreman will consider the following list of critical systems which may require assesment.

    a. ECCS systems
    b. AFW systems
    c. Safeguards Vent. systems
    d. Off site power availability
    e. Diesel Generators
    f. Instrument air systems
    g. Liquid and Gas Radwaste systems
    h. Containment Fan Coolers
    i. Component Cooling Water system
    j. Auxiliary Saltwater system
    k. Condenser and Circ. Water System
    l. Steam dump system
    m. Makeup water system

35. Continue to operate the ECCS systems in the Cold Leg Recirculation MODE for 19-1/2 hours, then perform Appendix B, Cold Leg Recirculation/Hot Leg Recirculation Changeover Procedure. Continue to operate in the Hot Leg Recirculation Mode until authorized by the Plant Superintendent to terminate hot leg recirculation.

APPENDIX A

LOCA INJECTION/RECIRCULATION CHANGEOVER PROCEDURE

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. All steps listed below must be carried out expeditiously, in a precise orderly sequence. Do not interrupt the operation until all actions are completed. When both ECCS trains are initially available and a valve fails to respond or to complete its demanded operations, postpone any corrective action until the subsequent operational steps are performed.	1. CAUTION: If a loss of off-site power has occurred in coincidence with the LOCA and all 3 diesel generators are running and supplying the vital busses, continue with these instructions as written. If a diesel generator has failed, go to Appendix E (Loss of Off-Site Power During LOCA with Loss of Diesel Generator) for additional guidance before proceeding
2. Monitor RWST level closely. If the RHR pumps do not trip at 33%, trip them manually.	
3. If during this operation the RWST approaches L-Lo level, stop all pumps taking suction from the RWST. Restart pumps after the RHR system is aligned to provide suction.	
4. At the vital 480 volt load centers F and H, close the breakers for the following valves if not already performed. 8980 RHR pumps supply from RWST 52-1F-31. 8976 SI pump supply from RWST 52-1H-20	4. This step may be delayed until time permits.
5. Reset SI, Phase B isolation and containment spray signals if not already reset.	

APPENDIX A (cont.)

ACTION	COMMENTS
6. If off-site power is available, shutdown all 3 diesel generators and leave them in Auto <u>if this has not been performed</u>.	
7. Close (cut in) the series contactors for the following valves from the toggle switches in the control room <u>if not already CUT IN</u>. 8809 A & B RHR pumps injection to cold legs. 8974 A & B SI pumps recirculation. 8982 A & B RHR pumps suction from sump.	
8. With the SI, charging and possibly the spray pumps still taking suction from the RWST and the RHR pumps tripped at 33% in the RWST proceed to transfer RHR suction to the containment sump as follows in steps 9 thru 29. Monitor RWST level, trip all pumps taking suction from the RWST upon reaching Lo-Lo Level in the tank. Restart pumps after RHR suction supplied.	
9. Check that the containment recirculation sump level indicators read at least 40% to provide adequate NPSH to the RHR pumps.	
10. Close the two RHR heat exchanger outlet crosstie valves (8716A & B).	
11. Close the No. 2 RHR pump normal suction valve (8700B).	

APPENDIX A (cont.)

ACTION	COMMENTS
12. Open the No. 2 RHR pump suction valve from the containment recirculation sump (8982B).	
13. Open the component cooling water outlet valve from No. 2 RHR heat exchanger (FCV-364.)	
14. Restart No. 2 RHR pump and check flow to the reactor vessel (VB-1).	
15. Close the s fety injection pump recirculation valves (8974A & B).	
16. Open the safety injection pump suction from RHR pump no. 2 (8804B). Check for increased safety injection pump flow and pressure.	
17. Close the safety injection pumps normal suction valve (8976) from the RWST.	17. This step may be delayed if breakers in step 4. have not been racked in.
18. Open the alternate suction valves for the centrifugal charging pumps (8807A & B). Check for increasing charging pump flow and pressure.	
19. Close the no. 1 RHR pump normal suction valve (8700A).	
20. Open the no. 1 RHR pump suction valve from the containment recirculation sump (8982A).	
21. Check the level in the recirculation sump again for adequate NPSH.	
22. Open the component cooling water outlet valve from no. 1 RHR heat exchanger (FCV-365).	

APPENDIX A (cont.)

ACTION	COMMENTS
23. Restart the no. 1 RHR pump and check for flow to the reactor vessel.	
24. Open the centrifugal charging pumps alternate suction valve from RHR pump no. 1 (8804A). Check for increased charging pump flow and pressure.	
25. Close the charging pumps normal suction valves from the RWST (8805A & B).	
26. Close the RHR pumps normal suction supply valve from the RWST (8980).	26. This step may be delayed if breakers in step 4. have not been racked in.
27. When the lo lo level point is reached in the RWST, trip both containment spray pumps.	27. Trip any other engineered safeguard pumps that are still taking suction from the RWST. This will only be necessary if the preceeding steps have not been completed prior to reaching the lo lo level set point.
28. If step 4, 17 and 26 were delayed, perform these steps at this time.	
29. If either RHR pump failed, either containment sump to RHR suction valve failed to open (valves 8982A or B) or either RHR Train to centrifugal charging pumps or SI pumps suction valves 8804A or B failed to open, go to Appendix D (RHR Train Failure).	
30. If W.R. RCS pressure is less than 1065 psig, return to step 25 of the LOCA procedure.	30. Large Break LOCA
If W.R. RCS pressure is greater than 1065 psig, return to LOCA procedure step that was left to perform this appendix.	Small Break LOCA

APPENDIX B

COLD LEG RECIRCULATION/HOT LEG RECIRCULATION CHANGE OVER PROCEDURE

ACTION	COMMENTS
Hot Leg Recirculation Phase	
1. Approximately 19-1/2 hours after the injection phase, the recirculation flow path is changed from the cold legs to the hot legs.	
2. It is important during this phase to insure that two separate and redundant trains of both RHR and safety injection pumps are established.	
3. Close the no. 1 RHR pump cold leg injection valve (8809A).	
4. Close the RHR pump no. 1 valve to the containment spray header (9003A) if it is open.	
5. Check open the RHR pump no. 1 valve to the safety injection pumps and centrifugal charging pumps (8804A).	
6. Shut down both centrifugal charging pumps and close both BIT inlet valves (8803A and B).	
7. Shutdown no. 1 safety injection pump.	
8. Close no. 1 safety injection pump discharge crosstie valve (8821A).	
9. Open the no. 1 safety injection pump hot leg injection valve (8802A).	
10. Start no. 1 safety injection pump and check for rated flow of approximately 650 gpm.	
11. Close the no. 1 safety injection pump normal suction valve (8923A).	

APPENDIX B (cont.)

ACTION	COMMENTS

12. Close the no. 2 RHR pump cold leg injection valve (8809B).

13. Check closed the no. 2 RHR pump valve to the containment spray header (9003B).

14. Check open the no. 2 RHR pump to the safety injection pumps alternate suction valve (8804B).

15. Shutdown no. 2 safety injection pump.

16. Close the no. 2 safety injection pump discharge crosstie valve (8821B).

17. Close the safety injection pumps common discharge valve (8835) to the cold legs.

18. Open the no. 2 safety injection pump hot leg injection valve (8802B).

19. Start the no. 2 safety injection pump and check for rated flow of approximately 650 gpm.

20. Close the no. 2 safety injection pump normal suction valve (8923B).

21. The systems are now set up in two completely separate and redundant trains with each RHR pump supplying a separate SI pump. No containment spray or direct RHR recirculation are utilized.

22. If desired, direct RHR recirculation to the hot legs may be used simultaneously as follows:

    a. Open the No. 1 RHR train discharge crosstie valve (8716A).

APPENDIX B (cont.)

ACTION	COMMENTS
b. Open the common RHR hot leg injection valve (8703).	
c. Adjust the RHR train No. 1 flow (HCV-638) to insure adequate suction to the SI pump and to maintain RHR pump motor current less than 57.5 amps.	
23. If containment spray is desired for any reason (such as loss of fan cooler unit) it may be accomplished as follows:	
a. Open the No. 2 RHR Train valve to the containment spray header (9003B).	
b. No flow control is available in this operation, so SI pump suction pressure and RHR pump motor current must be closely watched.	
24. Divide the auxiliary saltwater system into two separate trains as specified in Operating Procedure E-5.	24. If Appendix E was used to correct a failed diesel generator problem, do not separate the two trains.
25. Line up the component cooling water system for long term recirculation as specified in Operating Procedure F-2.	25. Same note as 24 above.

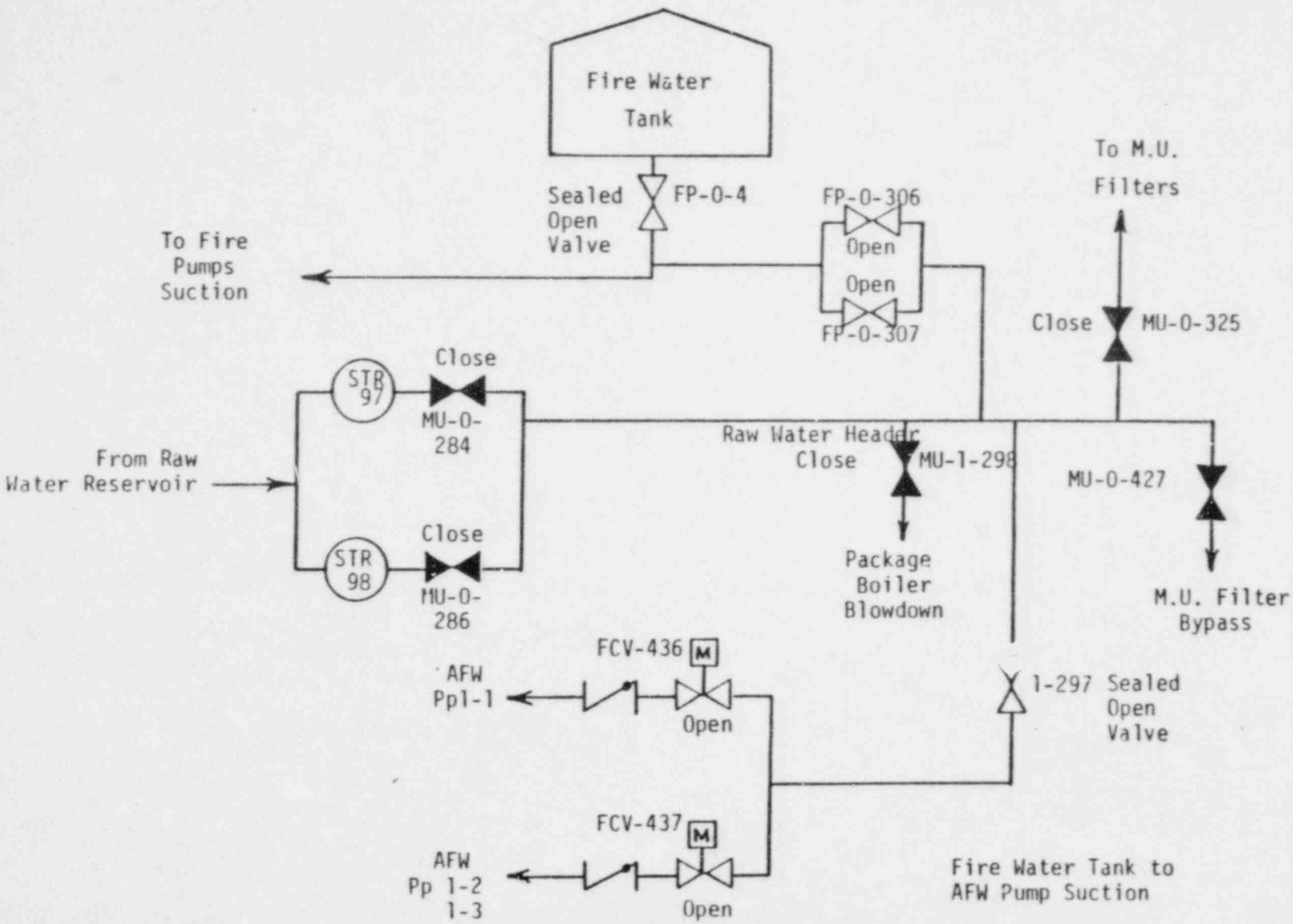
APPENDIX C

AUXILIARY FEED PUMP SUCTION SUPPLY FROM FIRE WATER TANK

The Operator has 20 minutes to perform this operation after the Lo Lo level alarm on the condensate storage tank and before the AFW pumps lose suction. This provides sufficient time; however, the operator must not delay and must carry out the valve line up in order as written.

If the AFW pumps are being supplied from the raw water reservoir and a seismic event occurs with resultant loss of AFW suction and auxiliary feedwater flow to the steam generators, the steam generators will boil dry in about 30 minutes. Under these conditions, it is especially important to expedite this procedure and reestablish AFW flow to the steam generators prior to the reactor losing its heat sink.

ACTION	COMMENTS
Using the attached drawing, proceed to supply the AFW pumps suction from the fire water tank.	
1. Close or verify closed MU-0-284 and MU-0-286.	1. Closing these valves prevents losing fire water out a possible break in the reservoir supply line.
2. Close or check closed MU-1-298.	
3. Close or check closed MU-0-325.	
4. Close or check closed MU-0-427.	
5. Open FP-0-306 and FP-0-307.	
6. Notify the control room that the suction for the AFW pumps is now available from the fire water tank.	
7. From the control room open FCV 436 and 437.	
8. Proceed to the auxiliary feedwater pumps and vent the pump casings if required to remove air.	



Fire Water Tank to AFW Pump Suction

APPENDIX D

RHR TRAIN/COMPONENT FAILURE

A. FAILURE IN RHR TRAIN NO. 2

ACTION	COMMENTS
1. If RHR pump No. 2 failed, containment sump to RHR valve 8982B or RHR pump No. 2 to SI suction valve 8804B failed to open proceed with steps a. to d. to provide charging and SI pump suction from RHR pump No. 1.	1. Monitor RWST level, when the level reaches the lo lo alarm setpoint, trip all safeguards pumps taking suction from the tank.
a. Verify RHR pump No. 1 crosstie valves to SI pump suction 8807A or B open.	
b. Close or verify closed Train No. 2 to SI pump suction valve 8804B.	
c. Throttle HCV 638 to ensure adequate suction for the SI and centrifugal charging pumps (approximately 20 psig above containment pressure) while maintaining RHR pump motor current less than 57.5 amps.	
d. Return to the LOCA procedure step that was left to perform Appendix A.	

APPENDIX D

B. FAILURE IN RHR TRAIN NO. 1

ACTION	COMMENTS
1. If RHR pump No. 1 failed, containment sumpe to RHR valve 8982A or RHR pump No. 1 to charging pumps suction valve 8804A failed to open, proceed with steps a. to d. to provide charging and SI pump suction from RHR pump No. 2.	1. Monitor RWST level, when the level reaches the lo lo alarm setpoint, trip all safeguards pumps taking suction from the tank.
a. Verify RHR pump No. 2 crosstie valves to SI pump suction 8807A or B open.	
b. Close or verify closed Train No. 1 to SI pump suction valve 8804A.	
c. Throttle HCV 637 to ensure adequate suction for the SI and centrifugal charging pumps (approximately 20 psig above containment pressure) while maintaining RHR pump motor current less than 57.5 amps.	
d. Return the the LOCA procedure step that was left to perform Appendix A.	

APPENDIX E

LOSS OF OFF-SITE POWER DURING LOCA WITH LOSS OF DIESEL GENERATOR

ACTION — COMMENTS

A. If a diesel generator fails during the LOCA procedure, continue with the LOCA procedure as written until Appendix A (LOCA Injection/Recirculation Change Over Procedure) is to be performed. Then follow the guidance given below for the alignment for cold leg and hot leg recirculation.

B. Diesel Generator Failure

Diesel Generator 1-1 Failure

1. If diesel generator 1-1 has failed, the following steps should be used to align the system for cold leg recirculation

   a. Verify the following pumps are running.

      ASW Pumps 1 and 2
      AFW Pump 3
      Charging Pumps 1 and 2
      CCW Pumps 1 and 2
      SI Pump 1
      C.S. Pump 1
      RHR Pump 1

   b. With diesel generator 1-1 failed, do the following steps in the order given below in Appendix A (LOCA Injection/Recirculation Change Over Procedure).

      Steps 2,3,4,5,7,8,9,10,15,18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, then do step 17.

APPENDIX E (Cont.)

ACTION	COMMENTS

c. Throttle HCV-638 to provide adequate suction for the SI and charging pumps (approximately 20 psig above containment pressure while maintaining RHR pump motor current less than 57.5 amps).

d. Verify the cold leg recirculation flow path as follows:

RHR pump 1 from containment sump to cold legs 1 and 2, SI pump 1 from RHR pump 1 to all cold legs, charging pumps 1 and 2 from RHR pump 1 to all cold legs via the BIT.

Containment spray pump 1 from RWST to spray headers.

2. Proceed as follows after 19-1/2 hours for hot leg recirculation.

a. Perform the following steps in the order given below in Appendix B (Cold Leg Recirculation/Hot Leg Recirculation Change Over Procedure).

Steps: 1,3,4,5,6,7,8,9,10,11.

b. Close HCV-638 RHR discharge valve.

c. Open 8716A RHR pump crosstie valve.

d. Open 8703 RHR to hot legs 1 and 2.

e. Throttle open HCV 638 to hold an RHR pressure greater than 29 psig above containment pressure or less than 57.5 amps on RHR pumps.

APPENDIX E (Cont.)

ACTION	COMMENTS

f. Verify hot leg recirculation flow path as follows:

RHR pump 1 from containment sump to SI pump 1 suction via valves 8804A, 8807A and B and to hot legs 1 and 2 via valve 8703.

SI pump 1 from RHR pump 1 to RCS hot legs 1 and 2

Diesel Generator 1-2 Failure

1. If diesel generator 1-2 has failed the following steps should be used to align the system for cold leg recirculation.

   a. Verify the following pumps are running.

      ASW Pumps 1
      AFW Pumps 2 and 3
      Charging Pump 1
      CCW Pumps 1 and 3
      SI Pumps 1 and 2
      C.S. Pump 2
      RHR Pump 2

   b. With diesel generator 1-2 failed, do the following steps in the order given below in Appendix A (LOCA Injection/Recirculation Change Over Procedure).

      Steps: 2,3,4,5,7,8,9,10,11,12
      13,14,15,16,17,18,25,26,27,28.

APPENDIX E (Cont.)

ACTION | COMMENTS

c. Throttle HCV-637 to provide adequate suction for the SI and charging pumps (approximately 20 psig above containment pressure while maintaining RHR pump motor current less than 57.5 amps).

d. Verify the cold leg recirculation flow path as follows:

Containment spray pump 2 from RWST to spray headers. RHR pump 2 from containment sump to cold legs 3 and 4, SI pumps 1 and 2 from RHR pump 2 to all cold legs, charging pump 1 from RHR pump 2 to all cold legs via the BIT.

2. Proceed as follows after 19-1/2 hours for hot leg recirculation.

   a. Perform the following steps in the order given below in Appendix B (Cold Leg Recirculation/Hot Leg Recirculation Change Over Procedure).

      Steps: 1,6,7,8,9,10,12,13,14, 15,16,18,19.

   b. Close 8807A and B (RHR Pump No. 2 to charging pumps).

   c. Close HCV-637 RHR discharge valve.

   d. Open 8716B RHR crosstie valve.

   e. Open 87C3 RHR to hot legs 1 and 2.

APPENDIX E (Cont.)

ACTION	COMMENTS

f. Throttle open HCV-637 to hold an RHR pressure greater than 20 psig above containment pressure or less than 57.5 amps on the RHR pumps.

g. Verify the hot leg recirculation flow path as follows:

RHR pump 2 from containment sump to SI pumps 1 and 2 suction, via valve 8804B and to hot legs 1 and 2 via 8703.

SI pumps 1 and 2 from RHR pump 2 to all RCS hot legs.

Diesel Generator 1-3 Failure

1. If diesel generator 1-3 has failed, the following steps should be used to align the system for cold leg recirculation.

   a. Verify the following pumps are running

      ASW Pump 2
      AFW Pump 2
      Charging Pump 2
      CCW Pumps 2 and 3
      SI Pump 2
      C.S. Pumps 1 and 2
      RHR Pumps 1 and 2

   b. With the diesel generator 1-3 failed, do the following steps in order given below in Appendix A (LOCA Injection/Recirculation Change Over Procedure).

      Steps: 2,3,4,5,7,8,9,10,11,12,13 14,15,16,17,18.19,20.21.22,23,24, 27,28.

APPENDIX E (Cont.)

ACTION	COMMENTS

c. Throttle HCV-638 and 637 to provide adequate suction for the SI and charging pump (approximately 20 psig above containment pressure while maintaining RHR pump motor current less than 57.5 amps).

d. Verify cold leg recirculation flow path as follows:

Containment spray pumps 1 and 2 from RWST to spray headers.
RHR pump No. 2 from containment sump to cold legs 3 and 4 and to SI pump No. 2.

SI pump No. 2 from RHR No. 2 to all cold legs.

RHR pump No. 1 from containment sump to cold leg 1 and 2 and to No. 2 centrifugal charging pump.

Centrifugal charging pump No. 2 from RHR Pump No. 2 to all cold legs via the BIT.

2. Proceed as follows after 19-1/2 hours for hot leg recirculation.

a. Perform the following steps in the order given in Appendix B (Cold Leg Recirculation/Hot Leg Recirculation Change Over Procedure).

Steps: 1,2,3,4,6,12,13,14,15,16,17,18,19,20, then perform step 22.

APPENDIX E (Cont.)

ACTION	COMMENTS
b. Close 8804A RHR supply to charging pumps.	b. All BIT inlet and outlet valves will not close with a failure of 1-3 diesel generator. This step prevents flow to the cold legs via the charging pumps and BIT.
c. Verify hot leg recirculation flow path as follows. RHR pump 1 from containment sump to hot legs 1 and 2 via HCV-638. RHR pump No. 2 from containment sump to SI pump No. 2 via 8804B. SI pump 2 from RHR pump 2 to RCS hot legs 3 and 4.	

APPENDIX F

DETERMINATION OF ADEQUATE CORE COOLING

This appendix provides the guidance to determine adequate core cooling if inadequate core cooling is suspected. Further, the instructions for regaining adequate core cooling is presented.

ACTION	COMMENTS
1. Monitor the core exit thermocouple temperatures. a. If the P-250 is available go to step 2. b. If the P-250 is not available go to step 3.	
2. If 5 or more P-250 thermocouple readings exceed 1200°F, notify the Shift Foreman that inadequate core cooling exists and go to step 5. If there are not 5 or more that exceed 1200°F, discontinue this appendix but continue to monitor the thermocouple readings.	
3. At the thermocouple incore board in the control room, Monitor 10 core-centered thermocouples. If any 3 of the 10 thermocouples exceed 700°F (Pegged hi) proceed to step 4. If there are not 3 thermocouples that exceed 700°F, discontinue this appendix but continue to monitor the thermocouple readings.	
4. If SI flow to the RCS and AFW flow to the steam generators cannot be verified, notify the Shift Foreman that inadequate core cooling may exist and go to step 5. If both SI slow to the RCS and AFW flow to the steam generators can be verified, discontinue this Appendix but continue to monitor the thermocouple readings.	
5. The Shift Foreman will verify if inadequate core cooling exists using the appropriate steps above. If inadequate core cooling exists the Shift Foreman will direct operations as follows:	

APPENDIX F (Cont.)

ACTION	COMMENT
a. Declare a General Emergency Implement the instructions given in Emergency Procedure G-1 regarding on and off-site protective actions.	
b. Attempt to establish SI flow to the RCS and AFW flow to the steam generators.	
c. Continue monitoring core outlet temperature to determine the effectiveness of the remaining actions.	
d. DEPRESSURIZE THE RCS by method 1 or 2 below.	
1) Dump steam to the condenser or atmosphere if the steam generator levels are in the narrow range and AFW flow is evident.	1) THIS IS THE PREFERRED METHOD.
2) Verify the SIS or charging pumps are running and available to deliver water to the RCS THEN Open the pressurizer PORV's.	2) Opening the PORV's will provide a drop in RCS pressure sufficient to allow the SI flow required to cool the core. This method is to be used if 1) (above) is ineffective.
e. If no means of depressurization are available, or if the depressurization did not result in decreasing core exit thermocouple temperatures, THEN START A RCP if possible. If the RCP fails after starting, replace the lost RCP with any remaining RCP.	e. Attempt to establish CCW and seal water flow to the pump; however, if CCW and/or seal water flow cannot be established, proceed to start a RCP. The pump must be started to move coolant thru the core.

APPENDIX G

BLACKOUT WITH SAFETY INJECTION EMERGENCY LOADING OF VITAL BUSSES

1. If the vital busses lose voltage prior to resetting the safety injection signal, the vital busses will automatically load the vital equipment given below. Verify the equipment has been loaded by observing breaker lights on the control board.

2. If the vital busses lose voltage after the safety injection signal has been reset, load or verify loaded the equipment given below onto the vital busses manually. Allow approximately 4 seconds between loading of each piece of equipment onto a given vital bus. Load or verify that the CFCU are running in low speed.

VITAL BUS F	VITAL BUS G	VITAL BUS H
D/G 1-3	D/G 1-2	D/G 1-1
MCC 1-F	MCC 1-G	MCC 1-H
CC Pp 1-1	CC Pp 1-2	SI Pp 1-2
SI Pp 1-1		
CFCU 1-2	RHR Pp 1-1	RHR Pp 1-2
CFCU 1-1	CFCU 1-3	CFCU 1-4
CCW Pp 1-1	CFCU 1-5	CCW Pp 1-3
ASW Pp 1-1	CCW Pp 1-2	AFW Pp 1-2
AFW Pp 1-1	ASW Pp 1-2	

3. Load the containment spray Pumps only if they were running prior to the blackout.

VITAL BUS G	VITAL BUS H
Cont Spray Pp 1-1	Cont Spray Pp 1-2

APPENDIX H

VERIFICATION OF ESF FUNCTIONING AND EFFECTIVENESS DURING LOCA

The primary function of the Shift Technical Adviser during any emergency is to evaluate plant conditions and advise the Shift Foreman on recovery procedures or actions to be taken to mitigate the consequences of the emergency. This appendix provides a method of evaluating the effectiveness of the ESF system during any LOCA and provides guidance on reclassifying the accident from a Site Emergency to a General Emergency should conditions warrant. The on shift Shift Technical Adviser (or his designated alternate if additional manpower is available) will make this evaluation as the event progresses and will make the necessary recommendations to the Shift Foreman based on his findings.

For all accidents except the LOCA[1], the maximum amount of activity which could be released is low enough so that persons located beyond the site boundary are not in immediate jeopardy even under the worst circumstances. In the case of the LOCA, however, it is theoretically possible that a large fraction of the core inventory of the volatile fission products (noble gases and iodines) could be released if one assumes multiple failures of the various engineered safety features. In such a case, doses within the LPZ could be very high, particularly to the thyroid gland. Loss of life is possible. As a result, it is imperative that downwind areas of the LPZ be evacuated as soon as possible in a LOCA situation where the functioning and/or effectiveness of the engineered safety feature systems is in doubt. In the case of the LOCA with inadequate core cooling, the major release would not be expected for at least two hours due to the time required to melt a large fraction of the core and the expected time before any containment failure would be likely. Thus, early recognition of the signs of ESF malfunction and prompt action in this event can prevent most, if not all, of the exposure of persons within the LPZ.

In the event of a LOCA, the Evaluations Coordinator should pay particular attention for signs that the ESF systems have malfunctioned, and if their effectiveness is in doubt, go ahead and classify the accident as a General Emergency and have the Emergency Coordinator notify those agencies given in General Appendix 2 of the emergency procedures and the NRC Operation Center immediately.

This Appendix and worksheets provide a checklist of three major categories of items to look for, and provides criterion to indicate that the various ESF have or have not functioned. In several cases there is more than one indicator to verify the functioning of a particular

[1]Which includes loss of the primary system heat sink and resultant expulsion of coolant from the primary system to the extent that safety injection is required.

APPENDIX H (Cont'd)

safety feature. For example, SI pump flow is measured directly by a flow indicator, but can also be inferred from pump operation and valve position indicators. If the various indications do not agree, a judgement must be made on the basis of what is considered to be the best evidence. Normally, however, the conservative course of action should be followed.

1. Worksheet for Verification of ESF Functioning and Effectiveness During a LOCA.

   This worksheet is to be used first to check that the minimum components in each of the ECCS and containment cooling systems have operated correctly. Although this does not guarantee that they will be effective, it provides reasonable assurance. Conversely, their failure vastly increases the chance of fuel melting or overheating and overpressurization of the containment (thereby jeopardizing its ability to retain fission products). Therefore, classify the accident as a General Emergency unless the minimum required components in each of the categories have functioned or are immediately initiated manually.

   The worksheet applies only to the injection phase. Although it is also necessary to maintain core cooling during the recirculation phase, the speed with which it must be restored is less cricical. However, if the ability to cool the core is lost, and is not likely to be restored rapidly, a General Emergency is justified.

   Included on the worksheet is the verification of containment isolation valve closure. Post-accident dose calculations are based upon the assumption that containment leakage rate is 0.1%/day. If the leakage rate exceeds this, the off-site does will increase proportionately. Although it is not possible to rapidly measure the leak rate, if all major penetrations that lead directly to the atmosphere are closed by at least one valve, the leak rate is probably reasonably low.

   If at least one valve in each of the paths on the worksheet (Page 39), Item 7, is not closed either automatically or by operator action within about 15 minutes, consider the accident a General Emergency.

2. Worksheet for Verification of Containment Pressure Less than Design Pressure

   This worksheet is used to determine if the internal containment pressure significantly exceeds the design pressure of the structure. Doubts are cast about the ability of the containment to retain

APPENDIX H (Cont'd)

the fission products (at least a maximum leakage rate of 0.1%/day) if design pressure is exceeded. It also indicates that the ESF are not functioning properly, since the pressure should never get that high if a minimum safeguards train works correctly.

The containment is designed for 47 psig coincident with the forces caused by a double design earthquake (0.4g). With no earthquake (or following completion of an earthquake), the containment can withstand 1.5 times the previous pressure, or 70 psig. Figure 1 on the worksheet shows the maximum allowable containment pressure as a function of the earthquake strength. If the curve is exceeded, the accident must be considered a General Emergency.

3. Worksheet for Verification of Containment Radiation Level Within Expectations

This worksheet is used to determine the extent of core damage. Two high range gamma instruments (RE-30 and RE-31) have been installed to measure the internal containment radiation level following a LOCA. Figure 1 on the worksheet shows the calculated response of these instruments as a function of time after a LOCA assuming various degrees of core damage. Any LOCA would be expected to release the activity contained in the coolant and so a reading of this magnitude should not be cause for concern. Similarly, in a major LOCA some cladding failure may occur due to the combined effects of DNB, depressurization, and metal-water reaction. Cladding failure will release some of the activity contained in the gap between the pellet and the cladding. However, the amount of cladding failure should be small. Thus, if the activity in the containment reaches the line labeled "100% gap release," it is indicative of gross core damage and/or the onset of fuel melting. In either case, the ECCS has failed to function as expected and the accident should be considered as a General Emergency. That is, if the containment radioactivity reaches or exceeds the value on Figure 1 for 100% gap release, consider the accident as a General Emergency.

If there is any question about the validity of the readings, or if the above instruments have failed, readings can be taken at the containment exterior with protable radiation protection equipment. Specifically, Figure 2 on the worksheet shows similar data for a CP type instrument located at the exterior surface of the steel liner (but inside the movable concrete shield) at the equipment hatch. Figure 3 on the worksheet shows the does rate outside the containment concrete wall. The readings at the equipment hatch can be made with an extension probe on a Victoreen

APPENDIX H (Cont'd)

Rad Gun. The HPI-1010 would be an appropriate instrument to use for measurements external to the concrete containment wall.

NOTE: Normal background is about 0.015 mR/hr on the HPI-1010.

WORKSHEET 1, APPENDIX H
WORKSHEET FOR VERIFICATION OF ESF FUNCTIONING
AND EFFECTIVENESS DURING A LOCA

PERFORMED BY ____________ DATE ________ TIME ________

1. BASIC ACCIDENT INFORMATION

   UNIT ______ DESCRIPTION ____________

   DATE/TIME OF OCCURENCE ________ / ________

2. CENTRIFUGAL CHARGING PUMPS

   a. Pump suction valve open ☐ 8805A ☐ 8805B
   b. BIT inlet valve open ☐ 8803A ☐ 8803B
   c. BIT outlet valve open ☐ 8801A ☐ 8801B
   d. Charging pump ON ☐ No. 1 ☐ No. 2
   e. Flow through FI-917 ________ gpm
   f. At least one train must be verified in service by verifying an operable flow path and the pump ON, or by verifying flow through FI-917. ☐ YES ☐ NO

   BY ____________ TIME ______

3. SAFETY INJECTION PUMPS

   a. Pump supply valve from RWST open ☐ 8976
   b. Pump suction valve open ☐ 8923A ☐ 8923B
   c. Pump discharge crosstie valve open ☐ 8821A ☐ 8821B
   d. Pump cold leg discharge open ☐ 8835
   e. SIS pump ON ☐ No. 1 ☐ No. 2
   f. RCS pressure 1500 psig ☐ YES ☐ NO psig ________
   g. Flow established FI-918 ______ gpm FI-922 ______ gpm
   h. If RCS 1500 psig, at least one train must be verified in service by verifying an operable flow path and the pump ON, or by verifying flow established through the FI. ☐ N/A ☐ YES ☐ NO

   BY ____________ TIME ______

4. ACCUMULATORS

   a. Accumulators discharge (LI's show low level) No. 1 ☐ LI950 or 951; No. 2 ☐ LI952 or 953; No. 3 ☐ LI954 or 955; No. 4 ☐ LI 956 or 957
   b. RCS pressure <600 psig? ☐ YES ☐ NO psig ________
   c. At least one LI per accumulator indicates discharge, if RCS <600 psig. ☐ N/A ☐ YES ☐ NO ☐

   BY ____________ TIME ______

WORKSHEET 1, APPENDIX H

5. RESIDUAL HEAT REMOVAL PUMPS

a. Pump supply valve from RWST open ☐ 8980

b. Pump suction valve open ☐ 8700A ☐ 8700B

c. HX outlet flow control valve open ☐ HDV-638 ☐ HCV-637

d. Cold leg injection valve open ☐ 8809A ☐ 8809B

e. RHR pump ON ☐ No. 1 ☐ No. 2

f. RCS pressure <170 psig ☐ YES ☐ NO psig ____________

g. Flow established ☐ FI-970 _____gpm ☐ FI-971 ____gpm

h. If RCS <170 psig, at least one train must be verified in service by verifying an operable flow path and the pump ON, or by verifying flow through the FI. ☐ N/A ☐ YES ☐ NO

BY ________________________ TIME __________

6. CONTAINMENT SPRAY AND COOLING

a. Spray additive tank valve open ☐ 8992

b. Spray additive tank valve open ☐ 8994A ☐ 8994B

c. Spray pump discharge valve open ☐ 9001A ☐ 9001B

d. Spray pump ON ☐ No. 1 ☐ No. 2

e. Fan coolers ON: No. 1 ☐; No. 2 ☐ ; No. 3 ☐ ; No. 4 ☐ ; No. 5 ☐

f. Containment Hi-Hi pressure signal received ☐ YES ☐ NO

g. At least two spray pumps have functioned OR one spray pump and three fan coolers functioned if a containment Hi-Hi pressure signal received.

☐ N/A ☐ YES ☐ NO

BY ________________________ TIME __________

7. CONTAINMENT ISOLATION VALVE CLOSURE

	INSIDE CONT.	OUTSIDE CONTAINMENT
a. Pressure/Vacuum relief closed	☐ FCV-662	☐ FCV-663 and 664
b. Purge supply closed	☐ FCV-660	☐ FCV-661
c. Purge exhaust closed	☐ RCV-11	☐ RCV-12

d. At least one valve in each of the above flow paths must be closed within 15 minutes ☐ N/A ☐ YES ☐ NO

BY ________________________ TIME __________

8. If item 2.f, 3.h, 4.c, 5.h, 6.g, OR 7.d, is NO, consider the accident as a General Emergency

BY ________________________ TIME __________

WORKSHEET 2, APPENDIX H
WORKSHEET FOR VERIFICATION OF CONTAINMENT PRESSURE
LESS THAN DESIGN PRESSURE

PERFORMED BY ______________________ DATE __________ TIME __________

1. BASIC ACCIDENT INFORMATION

UNIT _______ DESCRIPTION ______________________________

DATE/TIME OF OCCURRENCE ______________ / __________

2. EARTHQUAKE FORCE MONITOR (EFM-1)

EFM-1 Indicator	Peak Reading		Peak Earthquake Acceleration	
a. Longitudinal	_____ _____	x 0.01 =	______________	g
b. Vertical	__________	x 0.01 =	______________	g
c. Transverse	__________	x 0.01 =	______________	g

BY ______________________ TIME ______________

3. CONTAINMENT PRESSURE

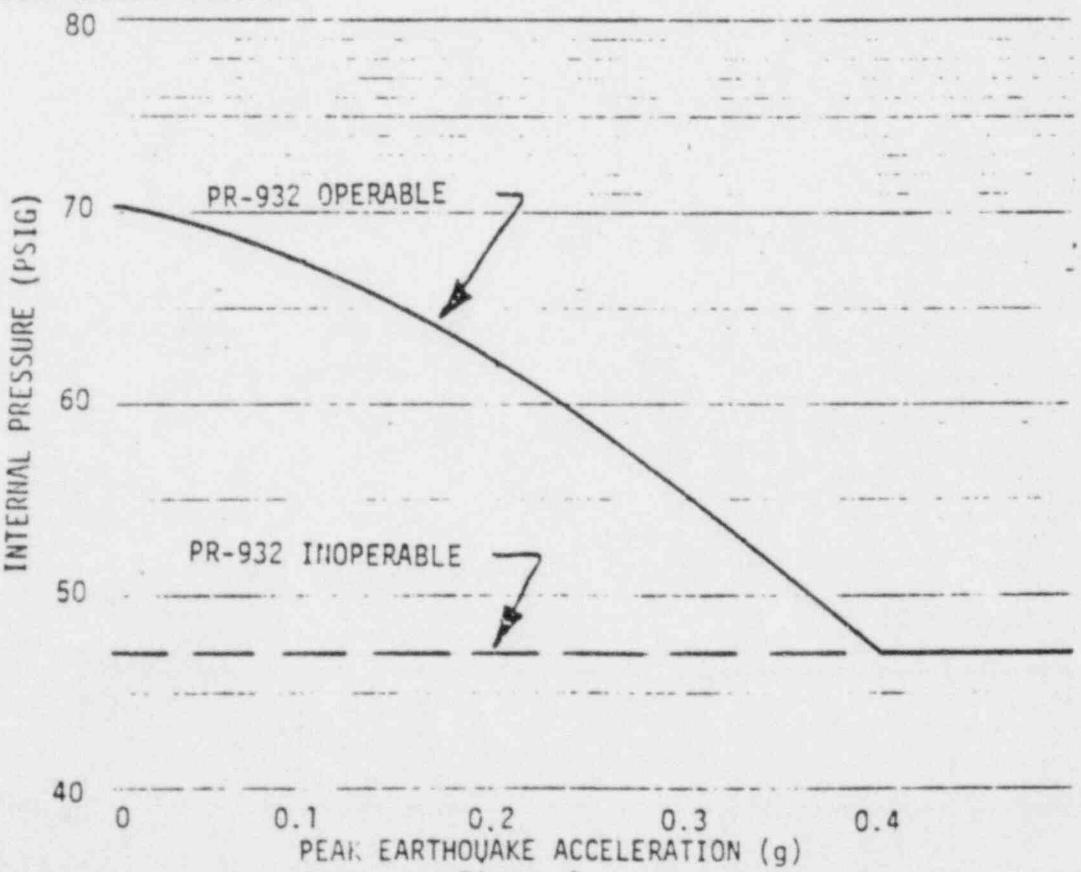


Figure 1
Internal Pressure Capability of the Containment
Versus Earthquake Acceleration

WORKSHEET 2, APPENDIX H

a. Using the largest acceleration in g's from the EFM-1 (item 2 above) and Figure 1 determine:

1) Maximum allowable pressure with PR-932 operable ____________ psig

2) Maximum allowable pressure without PR-932 operable ____________ psig

b. Using containment pressure indicator PR-932 or any 2 of 4 of the following indicators PI-934, 935, 936, 937, record containment pressure every two or three minutes for the first 15 minutes of the accident.

INSTRUMENT ID	TIME	PRESSURE (PSIG)

BY ______________________ TIME ________

4. DESIGN LIMITATION (complete 3.a OR 3.b)

a. If PR-932 was used in item 3.b to determine containment pressure, are any pressure readings greater than the maximum allowable pressure with PR-932 operable (item 3.a.1):

YES ☐ NO ☐

OR

b. If any 2 of 4 of the following indicators, PI-934, 935, 936, or 937, were used to determine containment pressure, are any pressure readings greater than the maximum allowable pressure without PR-932 operable (item 3.a.2):

YES ☐ NO ☐

5. If item 3.a OR 3.b is YES, consider the accident as a General Emergency.

BY ______________________ TIME ________

WORKSHEET 3, APPENDIX H
WORKSHEET FOR VERIFICATION OF CONTAINMENT RADIATION LEVELS

PERFORMED BY ______________________ DATE __________ TIME __________

1. BASIC ACCIDENT INFORMATION

   UNIT _____ DESCRIPTION ______________________________

   DATE/TIME OF OCCURRENCE ________________/______________

2. Using the lowest reading on one of the following, RE-30, RE-31, (in the Control Room), the Victoreen Rad Gun (between the equipment hatch steel door and movable concrete shield), or a HPI-1010 (at the containment concrete wall), determine the radiation level following an accident approximately every 15 minutes for 2 hours.

INSTRUMENT ID	TIME	DOSE RATE(R/HR)	BY

WORKSHEET 3, APPENDIX H

3. Determination of Expected Radiation Level

   a. Using Figure 1, (page 44) the time after the accident and readings from RE-30 and RE-31 (item 2), are radiation levels equal to or higher than expected for 100% gap release.

      ☐ N/A ☐ YES ☐ NO

   b. Using Figure 2, (page 45) the time after the accident and readings from the Victoreen Rad Gun (item 2), are radiation levels equal to or higher than expected for 100% gap release.

      ☐ N/A ☐ YES ☐ NO

   c. Using Figure 3, (page 46) the time after the accident and readings from the HPI-1010 (item 2), are radiation levels equal to or higher than expected for 100% gap release.

      ☐ N/A ☐ YES ☐ NO

4. If item 3.a., 3.b., OR 3.c. is YES, consider the accident as a General Emergency.

BY____________________________ TIME____________

WORKSHEET 3, APPENDIX H

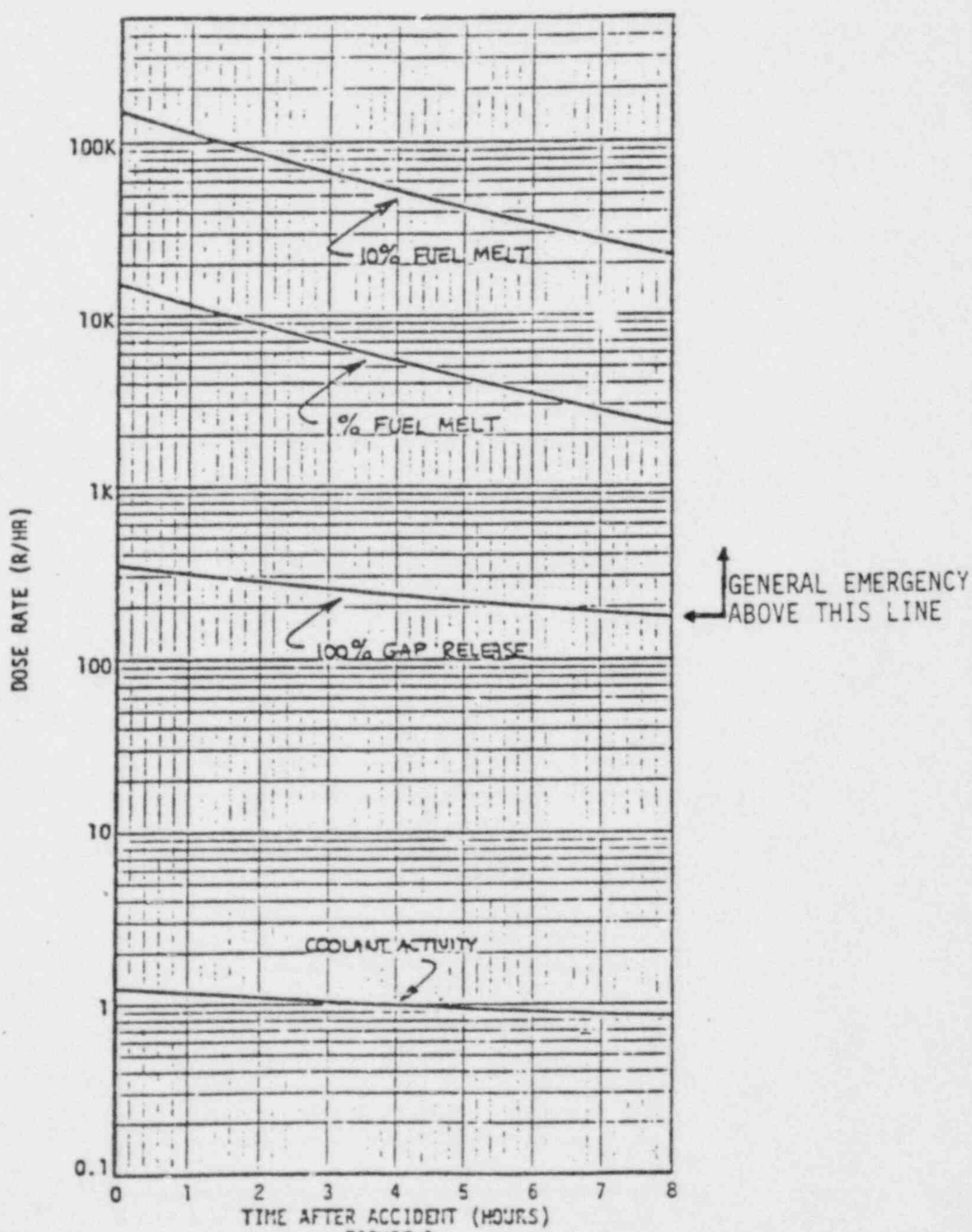


FIGURE 1

DOSE RATE INSIDE CONTAINMENT FOLLOWING LOCA

WORKSHEET 3, APPENDIX H

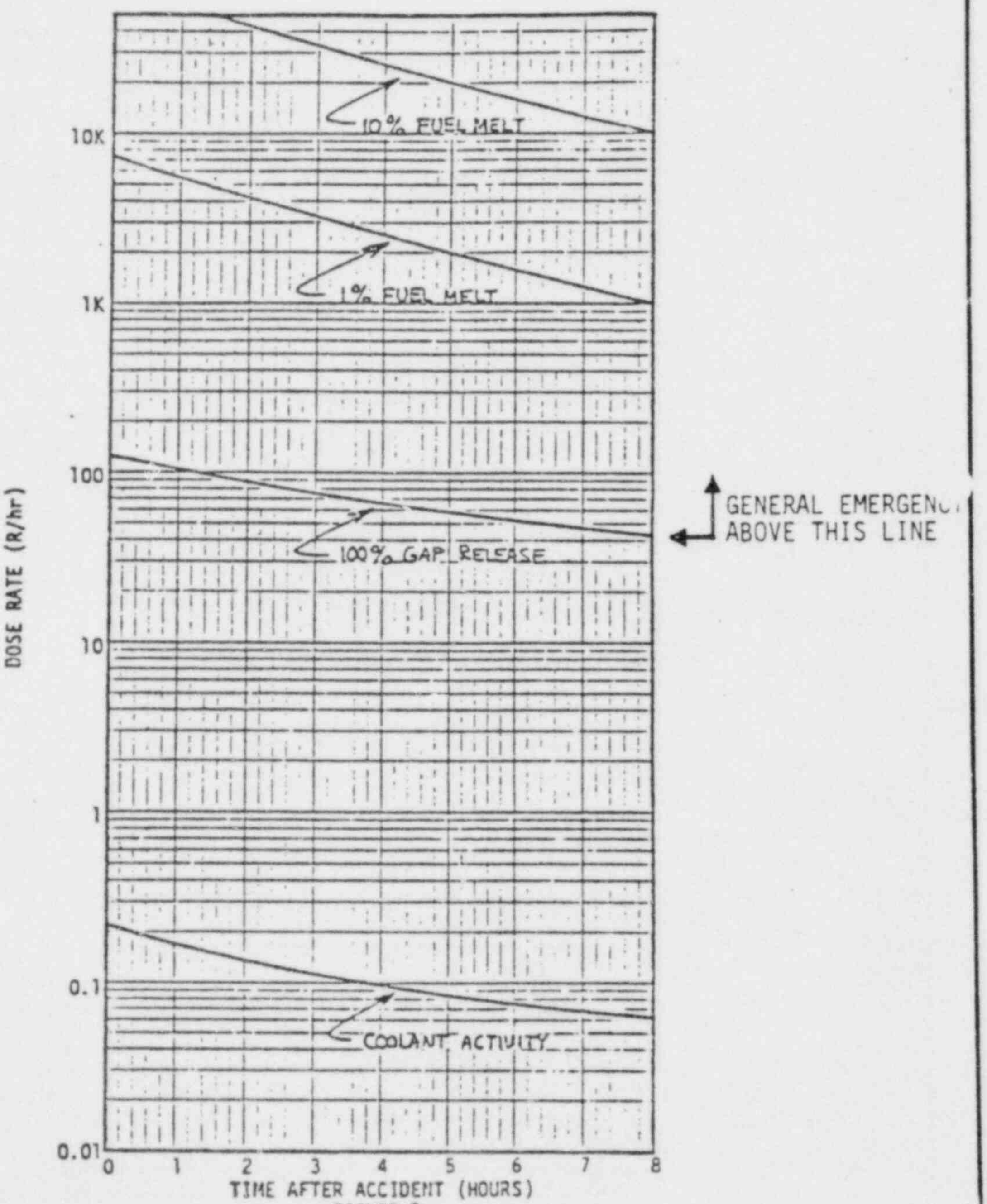


FIGURE 2

DOSE RATE OUTSIDE EQUIPMENT HATCH LINER FOLLOWING LOCA

WORKSHEET 3, APPENDIX H

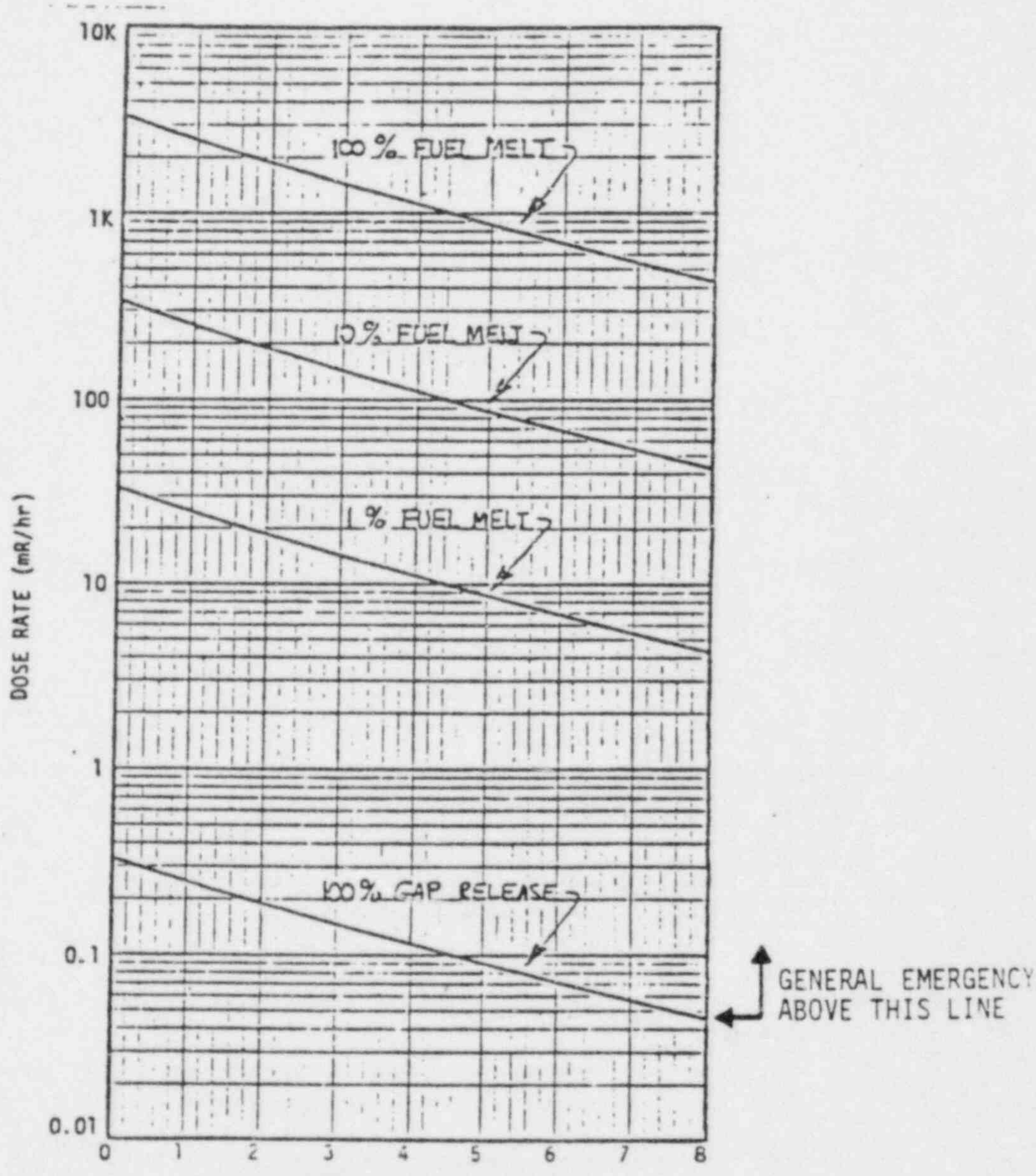
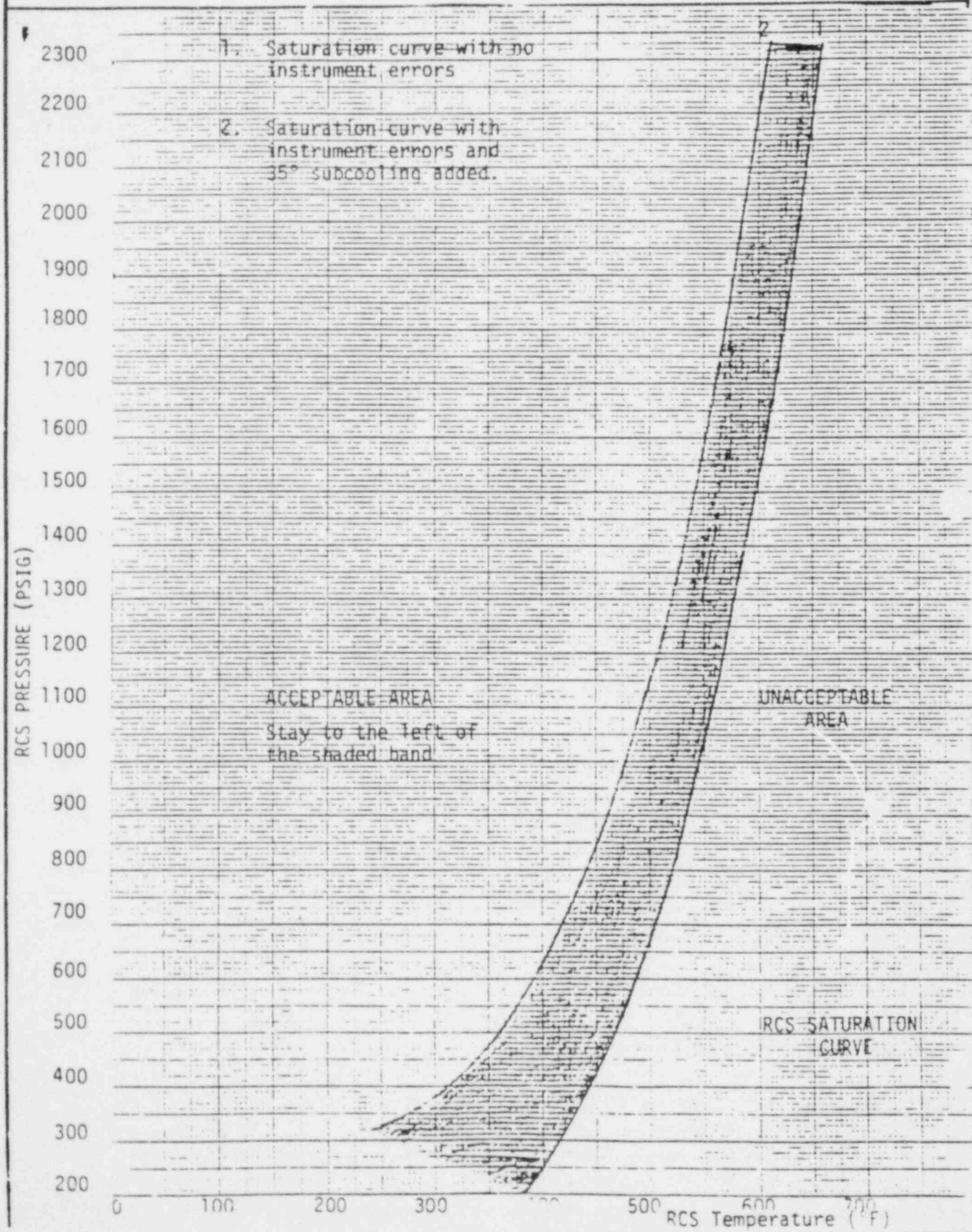


FIGURE 3
DOSE RATE OUTSIDE OF CONTAINMENT SHIELD WALL FOLLOWING LOCA

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2	NUMBER EP OP-1 REVISION 5 DATE 2/10/82 PAGE 48 OF 49
TITLE: LOSS OF COOLANT ACCIDENT	



APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Designate this event a Site Area Emergency. Notify plant staff and organizations required for this classification by Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organization" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

   b. In the event inadequate core cooling is verified per Appendix F or the verification of ESF Functioning and Effectiveness during LOCA, Appendix H, indicates ESF failure, reclassify this event as a General Emergency. Notify plant staff and response organizations in accordance with EP G-2 and EP G-3 and implement the instructions in EP G-1 regarding on and off-site protective actions.

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-2
REVISION 2
DATE 2/10/82

TITLE: EMERGENCY OPERATING PROCEDURE
LOSS OF SECONDARY COOLANT

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the operating steps to be taken in the event of a loss of Secondary Coolant. It is assumed that reactor trip and safety injection actuations have occurred. The operator should have already performed Emergency Operating Procedure No. OP-0.

SYMPTOMS

(See OP-0 Symptoms/Diagnostics)

AUTOMATIC ACTIONS

(See OP-0)

OBJECTIVES

1. To establish stablized reactor coolant system and steam generator conditions prior to plant cooldown.

2. To minimize the energy release due to the break by isolation of the break where possible.

3. To prevent the pressurizer safety valves from lifting by dumping steam from all steam generators to the main condenser when possible or to the atmosphere from the unaffected steam generators.

4. To isolate the auxiliary feedwater flow to the affected steam generator, to maximize auxiliary feedwater flow to the intact steam generators, and to minimize the energy release.

5. To borate the reactor coolant to establish and maintain reactor shutdown margin.

IMMEDIATE OPERATOR ACTIONS

1. Perform the immediate operator actions in the reactor trip with Safety Injection Emergency Procedure OP-0.

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Initiate the site Emergency alarm.	
2. If the pressurizer PORV's open at any time during this procedure, verify reclosure when the RCS pressure falls below the PORV setpoint. Isolate the PORV if the valve fails to close. If the valve remains open and cannot be isolated, go to OP-1.	
3. Verify main steam line isolation. If main steam line isolation has not occured, close the following valves.	3. Closing the valves helps identify the faulted steam generator.

Steam Generator No. 1

FCV-41 Steam line isolation valve

FCV-25 Isolation valve bypass

FCV-760 IC blowdown valve

Steam Generator No. 2

FCV-42 Steam line isolation valve

FCV-24 Isolation valve bypass

FCV-761 IC blowdown valve

Steam Generator No. 3

FCV-43 Steam line isolation valve

FCV-23 Isolation valve bypass

FCV-762 IC blowdown valve

Steam Generator No. 4

FCV-44 Steam line isolation valve

FCV-22 Isolation valve bypass

FCV-763 IC blowdown valve

ACTIONS	COMMENTS
4. Verify 10% atmospheric steam dumps open to hold steam generator pressures below the safety valves.	
5. If the RCS pressure stabilizes above the shutoff head of the RHR pumps, reset safety injection and shutdown both RHR pumps. Continue to monitor the RCS pressure and restart these pumps if the RCS pressure drops to the shutoff head of the pumps.	5. CAUTION: 1) Automatic reinitiation of safety injection will not occur after this step since the reactor trip breakers are open. If the operator has indication that an SI is required after this step, he must initiate it manually. CAUTION: 2) If loss of off-site power occurs after resetting safety injection, it will be necessary to load the safeguards equipment onto the vital busses manually. If safety injection is reinitiated manually after the loss of off-site power, the vital busses will automatically sequentially load the safeguard equipment. If loss of off-site power occurs, go to Appendix E (Blackout with SI Emergency Loading of Diesel Generators).
6. If W.R. RCS pressure decays below 1220 psig or is below 1220 psig and stable.	6. NOTE: The conditions for stopping RCP must be continuously monitored throughout the transient.
a. Again verify a minimum of one charging pump delivering flow and one SI pump delivering flow to the RCS,	
b. THEN STOP all four reactor coolant pumps. Maintain seal water flow to the RCP seals.	
c. Close the Centrifugal Charging pump recirculation valves (8105 & 8106).	c. If the RCS pressure is restored above 2000 psig reopen 8105 & 8106 to protect the pumps from damage.
d. If component cooling water to the RCP's is isolated due to a containment phase B isolation, stop all RCP's within 5 minutes and maintain seal flow as above.	

ACTIONS	COMMENTS
7. Monitor steam generator pressures to determine the faulted steam generator. If one steam generator has a low steam generator pressure, terminate AFW flow to that steam generator.	7. The low pressure steam generator is the faulted steam generator.
Steam Generator No. 1 Low Pressure	
Close LCV's 106 and 110.	
Steam Generator No. 2 Low Pressure	
Close LCV's 107 and 111.	
Steam Generator No. 3 Low Pressure	
Close LCV's 108 and 115.	
Steam Generator No. 4 Low Pressure	
Close LCV's 109 and 113.	
If all steam generators are depressurized or depressurizing, maintain AFW flow to all steam generators until the faulted steam generator is identified.	
8. If all steam generator pressures are stable.	
a. Dispatch an operator to inspect the feedwater and main steam lines for a possible break. If a break is found, continue with step 10.	a. AFW flow to the individual steam generators may supply additional information as to the break location.
9. If a break is not found, return to Accident Diagnostics step 3 in OP-C.	

ACTIONS	COMMENTS
*10. Verify AFW flow to all nonfaulted steam generators. Maintain maximum AFW flow until steam generator narrow range level indicators are reading on scale.	
11. If steam generator water level increases in an unexpected manner in one steam generator, go to OP-3A steam generator tube rupture.	
12. Monitor the condensate storage tank and upon reaching approximately 10% level, perform a. or b. below.	12. If the lo lo level alarm occurs on the CST, the operator has approximately 25 minutes to perform items a. or b.

a. Verify a level in the raw water storage reservoir; then open FCV-436 and FCV-437 (Reservoir supply to AFW pumps). Allow the AFW pumps to run during the transfer. Monitor the AFW flow closely. If AFW flow is lost, trip all 3 AFW pumps until the transfer is complete, then restart the pumps.

b. If the raw water storage reservoir is not available, go to Appendix C (AFW Pump suction Supply from Fire Water Tank Procedure). Allow the AFW pumps to run during the transfer. Monitor the AFW flow closely. If AFW flow is lost, trip all 3 AFW pumps until the transfer is complete, then restart the pumps.

13. If containment spray has been initiated and containment pressure has decreased to less than 22 psig.

a. Reset Train A and Train B containment spray actuation.

b. Verify all CFCU's running on low speed.

c. Stop both C.S. pumps.

d. Close spray additive valves.

14. If containment spray has been initiated and containment pressure remains above 22 psig, reset containment spray signal and cancel the spray additive. Close valves 8994A&B. Monitor RWST level throughout this procedure and when the RWST low level alarm 33' is annunciated, perform a. to d. below.

ACTIONS	COMMENTS
a. Verify RHR pumps trip if they are running.	
b. Perform Appendix A (SI Injection/ Recirculation Changeover Procedure).	
c. Continue to spray with the C.S. pumps <u>without</u> additive until the RWST lo lo level alarm is annunciated, then shutdown the containment spray pumps.	
d. After the changeover to cold leg recirculation, use the RHR to continue to spray containment until containment pressure is less than 22 psig. Verify CFCU's running on lo speed prior to terminating containment spray.	
15. Terminate safety injection if criteria A, B or C below can be met. Monitor and stay within the Technical Specifications heatup/cooldown curves while trying to meet the criteria for terminating SI.	15. Conditions for termination should be continuously monitored throughout these instructions. If all steam generators are depressurized or depressurizing, do NOT terminate SI until the faulted steam generator is identified.
A.	
1) <u>One</u> wide range RCS loop T_{HOT} is $\leq$ 350°F. If possible, confirm by core exit thermocouples.	1) Attempt to maintain temperature <350°F. If the criteria described in "A" are used for termination of safety injection and the reactor coolant temperatures increase to >350°F, maintain the safety injection pumps in operation until all criteria for "B" or "C" below are satisfied.
2) <u>AND</u> wide range RCS pressure $\geq$700 psig and stable or increasing,	
3) <u>AND</u> PZR level >22% and rising,	
4) <u>AND</u> subcooled margin meter reading >35°F subcooled,	4) If the subcooling margin monitor becomes INOPERABLE or suspect, use the saturation curve to determine subcooling.

ACTIONS

5) AND auxiliary feedwater flow has been isolated to all depressurized steam generators and a) or b) is satisfied below.

   a) AFW flow to the unfaulted steam generators is greater than 205 gpm per steam generator,

   b) OR one steam generator narrow range water level is greater than 10%.

B.

1) ALL wide range RCS loop T_{HOT} are $\geq 350°F$,

2) AND W.R. RCS pressure ≥ 2000 psig stable or increasing,

3) AND PZR level >22%,

4) AND subcooled margin meter reading >35°F subcooled,

5) AND auxiliary feedwater flow has been isolated to all depressurized steam generators and a) or b) is satisfied below.

   a) AFW flow to the unfaulted steam generators is greater than 205 gpm per steam generator

   b) OR one steam generator narrow range water level is greater than 10%

6) AND containment pressure and containment radiation and containment recirc sump DO NOT exhibit abnormally high or increasing readings. If containment conditions are increasing, continue SI until criteria C can be met.

ACTIONS	COMMENTS
C.	
1) All wide range RCS loop T_{HOT} are $\geq$350°F,	
2) AND W.R. RCS pressure $\geq$2000 psig stable or increasing,	
3) AND PZR level >50%,	
4) AND subcooled margin meter reading >35°F subcooled,	
5) AND auxiliary feedwater flow has been isolated to all depressurized steam generators and a) or b) is satisfied below.	
a) AFW flow to the unfaulted steam generators is greater than 205 gpm per steam generator	
b) OR one steam generator narrow range water level is greater than 10%	
6) AND containment pressure or containment radiation or containment recirc. sump exhibit abnormally high or increasing readings.	6) If all steam generators are depressurized or depressurizing, do NOT terminate SI until the faulted steam generator is identified.

ACTIONS	COMMENTS
16. If criteria A,B or C above is met and SI has not been reset, reset the SI signal and stop the RHR and SI pumps. Stop one charging pump at a time and evaluated RCS pressure. Maintain sufficient charging flow to supply adequate seal injection flow.	16. CAUTION: 1) Automatic reinitiation of safety injection will not occur after this step since the reactor trip breakers are open. If the operator has indication that a SI is required after this step, he must initiate it manually.
17. Continue to monitor the RCS conditions.	
a. IF, RCS W.R. pressure decreases by 200 psig,	
b. OR, PZR level decreases by 10% unexpectedly from SI termination point,	b. NOTE: PZR level and press. are expected to drop when letdown is reestablished.
c. OR, indicated subcooling $< 35°F$,	
d. THEN, manually reinitiate SI pump operation to maintain RCS pressure and PZR level and return to Accident Diagnostic Section of OP-0.	
e. Safety injection may be terminated after the restart when reactor coolant pressure is being controlled to the nominal value which exists when safety injection was initially terminated ($T_H < 350°F$) or to a value greater than or equal to 2000 psig ($T_H \geq 350°F$) and when the reactor coolant indicated subcooling is greater than 50°F.	
18. If RCS conditions remain stable perform the following steps.	
a. Reset containment isolation phase A.	
b. If RCP's have been stopped, and W.R. RCS pressure is greater than 1220 psig, establish conditions for starting RCP's and start at least one RCP associated with an unfaulted steam generator.	b. Start pump 1-1 or 1-2 if associated with an unfaulted loop so that PZR spray is effective.
c. Verify AC turbine bearing oil pump and Hi pressure seal oil backup pump running after oil pressure decays to 11 psig and turbine bearing lift pump starts at 600 RPM turbine speed.	NOTE: If the Centrifugal Charging pump recirc valves (8105 & 8106) were closed in Step 6, reopen these valves if RCS pressure is restored above 2000 psig.

ACTIONS COMMENTS

d. Establish normal charging.

1) Open instrument air valves FCV-584 & 682.

2) Check open normal charging valve 8146

3) Close or check closed charging to auxiliary spray valve 8145.

4) Open charging line isolation valves MO-8107 & 8108.

5) Adjust HCV-142 & FCV-128 or reciprocal charging pump speed to achieve RCP seal flow and charging flow as required to maintain pressurizer level greater than 22%.

6) Open RCP seal return valves, MO-8100 & 8112. Check seal flow normal.

7) Check open or open centrifugal charging pump recirculation valves.

e. Establish normal letdown.

1) Open letdown valves LCV-459 & 460.

2) Open letdown isolation valve 8152.

3) Open one 75 gpm letdown orifice valve.

4) Verify PCV-135 opening by observing letdown flow.

f. Close the BIT inlet and outlet valves MO 8803A & B, and MO 8801 A & B.

g. Establish VCT makeup and transfer charging pumps suction to VCT.

1) Adjust VCT makeup blend to the no-load concentration.

2) Open VCT outlet valves LCV-112B and C.

3) Close RWST to charging pump suction valves 8805A & B.

ACTIONS	COMMENTS
4) Verify divert valve LCV-112A in AUTO.	
h. Verify or establish normal charging to maintain PZR level at 22% and W.R. RCS pressure at the value reached when SI was terminated or 2000 psig. Reestablish PZR heaters and return to automatic level and pressure control only after containment conditions are such that proper operation of the control systems can be verified.	
i. Stop all 3 diesel generators if off-site power available and place diesel generator control switches in AUTO.	
j. Place the main and feedpump turbines on turning gear at 0 RPM speed.	
19. Borate the RCS to the cold Xe free condition, if required.	
20. If the nonfaulted steam generator levels have returned to the narrow range regulate AFW flow to maintain levels in the narrow range.	
21. Verify 50°F subcooling on the RCS subcooling meter. If 50°F subcooling is present, go to step 24.	
22. If 50°F subcooling is not present, attempt to establish 50°F subcooling with method a. or b. below.	
a. <u>Dump Steam to the Condenser</u>. If the break is inside the MSIV's or if a feedwater line break was found, open the M.S. isolation valves from the <u>unfaulted</u> steam generators. Reset the steam dump control to the steam header pressure control mode. Lower the control setpoint to establish 50°F subcooling while maintaining less than 50°F HR RCS cooldown rate.	a. This is the preferred method. Safety injection pump operation should be reinitiated if an uncontrolled drop in RCS pressure or pressurizer level occurs during the cooldown process.

ACTIONS	COMMENTS
	During the controlled cooldown the low RCS pressure criteria for tripping RCP's will not apply.
b. Dump Steam to the Atmosphere. If the break is outside the main steam isolation valves, lower the setpoint on the 10% atmospheric steam dump valves from the unfaulted steam generators to establish 50°F subcooling while maintaining <50°F/HR RCS cooldown rate.	b. The steam generator blowdown radioactive levels should be checked prior to dumping steam to the atmosphere. If hi radiation is found in a steam generator use the remaining unfaulted steam generators to cool the plant. However, if no other method of cooldown is possible, dump steam to the atmosphere regardless of the radiation levels in the steam generator blowdown.
23. If 50°F subcooling cannot be achieved, reinitiate safety injection and go to OP-0 Diagnostics and reevaluate the accident	
24. If RCP's are not running, establish conditions for starting the pumps and start at least one reactor coolant pump not associated with the faulted steam generators.	24. The lo RCS pressure RCP trip criteria will not apply during the cooldown. Start RCP's 1 or 2 if possible, so that pressurizer spray is possible.
25. If 50°F subcooling is present and can be maintained, proceed with a normal cooldown using Operating Procedure L-5.	
26. Reinitiate SI if an uncontrolled RCS depressurization or PZR level decreases during the plant cooldown.	

APPENDIX A

SI INJECTION/RECIRCULATION CHANGE OVER PROCEDURE

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. All steps listed below must be carried out expeditiously, in a precise orderly sequence. Do not interrupt the operation until all actions are completed. When both ECCS trains are initially available and a valve fails to respond or to complete its demanded operation, postpone any corrective action until the subsequent operational steps are performed.	1. CAUTION: If a loss of off-site power has occurred in coincidence with this procedure and all 3 diesel generators are running and supplying the vital busses, continue with these instructions as written. If a diesel generator has failed, go to Appendix B (Loss of Off-site Power with Loss of Diesel Generator) for additional guidance before proceeding.
2. Monitor RWST level closely. If the RHR pumps do not trip at approx. 33%, trip them manually.	
3. If during this operation the RWST approaches a 0% level, stop all pumps taking suction from the RWST. Restart pumps after the RHR system is aligned to provide suction.	
4. At the vital 480 volt load centers F and H, close the breakers for the following valves if not already performed. 8980 RHR pumps supply from RWST 52-1F-31. 8976 SI pumps supply from RWST 52-1H-20.	4. This step may be delayed until time permits.
5. Reset SI and Phase B isolation signals if not already reset.	
6. If off-site power is available, shutdown all 3 diesel generators and leave them in AUTO if this has not been performed.	

APPENDIX A (cont)

ACTIONS	COMMENTS
7. Close (cut in) the series contactors for the following valves from the toggle switches in the control room if not already CUT IN.	
8809 A & B RHR pumps injection to cold legs.	
8974 A & B SI pumps recirculation.	
8982 A & B RHR pumps suction from sump.	
8. With the SI, charging and possibly the spray pumps still taking suction from the RWST and the RHR pumps tripped at 33% in the RWST, proceed to transfer RHR suction to the containment sump as follows in steps 9 thru 29.	8. Monitor RWST level, trip all pumps taking suction from the RWST upon reaching 0 % level in the tank. Restart pumps after RHR suction supplied.
9. Check that the containment recirculation sump level indicators read at least 40% to provide adequate NPSH to the RHR pumps.	
10. Close the two RHR heat exchanger outlet crosstie valves (8716A & B).	
11. Close the No. 2 RHR pump normal suction valve (8700B).	
12. Open the No. 2 RHR pump suction valve from the containment recirculation sump (8982B).	
13. Open the component cooling water outlet valve from No. 2 RHR heat exchanger (FCV-364).	
14. Restart No. 2 RHR pump and check flow to the vessel.	
15. Close the safety injection pump recirculation valves (8974A & B).	

APPENDIX A (cont)

ACTIONS	COMMENTS
16. Open the safety injection pump suction from RHR pump No. 2 (8804B). Check for increased safety injection pump flow and pressure.	
17. Close the safety injection pumps normal suction valve (8976) from the RWST.	17. This step may be delayed if breakers in step 4 have not been racked in.
18. Open the alternate suction vales for the centrifugal charging pumps (8807A & B).	
19. Close the No. 1 RHR pump normal suction valve (8700A).	
20. Open the No. 1 RHR pump suction valve from the containment recirculation sump (8982A).	
21. Check the level in the recirculation sump again for adequate NPSH.	
22. Open the component cooling water outlet valve from No. 1 RHR heat exchanger (FCV-365).	
23. Restart the No. 1 RHR pump and check for flow to the vessel.	
24. Open the centrifugal charging pumps alternate suction valve from RHR pump No. 1 (8804A). Check for increased charging pump flow and pressure.	
25. Close the charging pumps normal suction valves from the RWST (8805A & B).	
26. Close the RHR pumps normal supply valve from the RWST (8980).	26. This step may be delayed if breakers in step 4 have not been racked in.

APPENDIX A (cont)

ACTIONS	COMMENTS
27. When the lo lo level point is reached in the RWST, trip both containment spray pumps.	27. Trip any other engineered safeguard pumps that are still taking suction from the RWST. This will only be necessary if the preceding steps have not been completed prior to reaching the lo lo level setpoint.
28. If steps 4, 17 and 26 were delayed, perform these steps at this time.	
29. If either RHR pump failed, either containment sump to RHR suction valve failed to open (valves 8982A or B) or either RHR Train to centrifugal charging pumps or SI pumps suction valves 8804A or B failed to open, go to Appendix D (RHR Train Failure).	
30. Return to the procedure step that was left to perform this Appendix.	

APPENDIX B

LOSS OF OFF-SITE POWER DURING LOCA WITH LOSS OF DIESEL GENERATOR

ACTIONS | COMMENTS

A. If a diesel generator fails during this procedure, continue with the procedure as written until Appendix A (SI Injection/Recirculation Change Over Procedure) is to be performed. Then follow the guidance given below for the alignment for cold leg and hot leg recirculation.

B. Diesel Generator Failure

Diesel Generator 1-1 Failure

1. If diesel generator 1-1 has failed, the following steps should be used to align the system for cold leg recirculation.

   a. Verify the following pumps are running.

      ASW Pumps 1 and 2
      AFW Pump 3
      Charging Pumps 1 and 2
      CCW Pumps 1 and 2
      SI Pump 1
      C.S. Pump 1
      RHR Pump 1

   b. With diesel generator 1-1 failed, do the following steps in the order given below in Appendix A (SI Injection/Recirculation Change Over Procedure).

      Steps: 2,3,4,5,7,8,9,10,15,18, 19,20,21,22,23,24,25,26,27,28 then do step 17.

APPENDIX B (cont)

ACTIONS | COMMENTS

c. Throttle HCV-638 to provide adequate suction for the SI and charging pumps (approximately 20 psig above containment pressure while maintaining RHR pump motor current less than 57.5 amps).

d. The cold leg recirculation flow path would be as follows.

RHR pump 1 from containment sump to cold legs 1 and 2, SI pump 1 from RHR pump 1 to all cold legs, charging pump 1 and 2 from RHR pump 1 to all cold legs via the BIT.

Containment spray pump 1 from RWST to spray headers.

Diesel Generator 1-2 Failure

1. If diesel generator 1-2 has failed the following steps should be used to align the system for cold leg recirculation.

a. Verify the following pumps are running.

ASW Pumps 1
AFW Pumps 2 and 3
Charging Pump 1
CCW Pumps 1 and 3
SI Pumps 1 and 2
C.S. Pump 2
RHR Pump 2

APPENDIX B (cont)

ACTIONS | COMMENTS

b. With diesel generator 1-2 failed, do the following steps in the order given below in Appendix A (SI Injection/Recirculation Change Over Procedure).

Steps: 2,3,4,5,7,8,9,10,11,12, 13,14,15,16,17,18,25,26,27,28.

c. Throttle HCV-637 to provide adequate suction for the SI and charging pumps (approximately 20 psig above containment pressure while maintaining RHR pump motor current less than 57.5 amps).

d. The cold leg recirculation flow path would be as follows.

Containment spray pump 2 from RWST to spray headers. RHR pump 2 from containment sump to cold legs 3 and 4, SI pumps 1 and 2 from RHR pump 2 to all cold legs, charging pump 1 from RHR pump 2 to all cold legs via the BIT.

Diesel Generator 1-3 Failure

1. If diesel generator 1-3 has failed, the following steps should be used to align the system for cold leg recirculation.

   a. Verify the following pumps are running.

      ASW Pump 2
      AFW Pump 2
      Charging Pump 2
      CCW Pumps 2 and 3
      SI Pump 2
      C.S. Pumps 1 and 2
      RHR Pumps 1 and 2

APPENDIX B (cont)

ACTIONS | COMMENTS

b. With the diesel generator 1-3 failed, do the following steps in the order given below in Appendix A (SI Injection/Recirculation Change Over Procedure).

Steps: 2,3,4,5,7,8,9,10,11,12,13, 14,15,16,17,18,19,20,21,22,23,24, 27,28.

c. Throttle HCV-63[illegible] and 637 to provide adequate suction for the SI and charging pump (approximately 20 psig above containment pressure while maintaining RHR pump motor current less than 57.5 amps).

d. The cold leg recirculation flow path would be as follows.

Containment spray pumps 1 and 2 from RWST to spray headers. RHR pump No. 2 from containment sump to cold legs 3 and 4 and to SI pump No. 2.

SI pump No. 2 from RHR No. 2 to cold legs 3 and 4.

RHR pump No. 1 from containment sump to cold leg 1 and 2 and to No. 2 centrifugal charging pump.

Centrifugal charging pump No. 2 from RHR Pump No. 2 to all cold legs via the BIT.

APPENDIX C

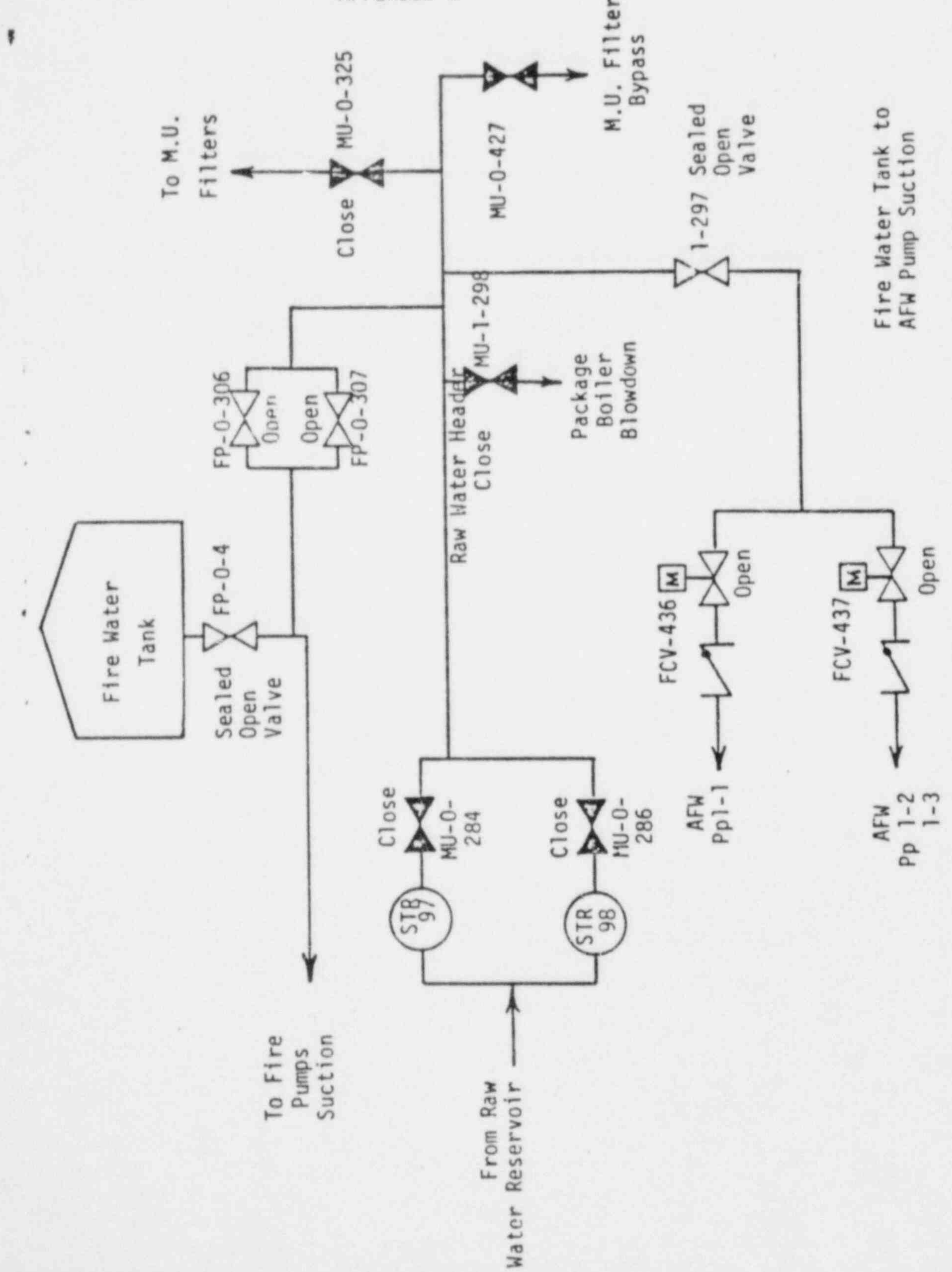
AUXILIARY FEED PUMP SUCTION SUPPLY FROM FIRE WATER TANK

The operator has 20 minutes to perform this operation after the lo lo level alarm on the condensate storage tank and before the AFW pumps lose suction. This provides sufficient time; however,the operator must not delay and must carry out the valve line up in order as written.

If the AFW pumps are being supplied from the raw water reservoir and a seismic event occurs with resultant loss of AFW suction and auxiliary feedwater flow to the steam generators, the steam generators will boil dry in about 30 minutes. Under these conditions, it is especially important to expedite this procedure and reestablish AFW flow to the steam generators prior to the reactor losing its heat sink.

ACTIONS	COMMENTS
Using the attached drawing, proceed to supply the AFW pumps suction from the fire water tank.	
1. Close or verify closed MU-0-284 and MU-0-286.	1. Closing these valves prevents losing fire water out a possible break in the reservoir supply line.
2. Close or check closed MU-1-298.	
3. Close or check closed MU-0-325.	
4. Close or check closed MU-0-427.	
5. Open FP-0-306 and FP-0-307.	
6. Notify the control room that the suction for the AFW pumps is now available from the fire water tank.	
7. From the control room open FCV-436 and 437.	
8. Proceed to the auxiliary feedwater pumps and vent the pump casings if required to remove air.	

APPENDIX C


To M.U. Filters
Close
MU-0-325
MU-0-427
M.U. Filter Bypass
1-297 Sealed Open Valve
Fire Water Tank to AFW Pump Suction
MU-1-298
Package Boiler Blowdown
Raw Water Header Close
FP-0-306
Open
Open
FP-0-307
Fire Water Tank
FP-0-4
Sealed Open Valve
FCV-436
M
Open
FCV-437
M
Open
AFW Pp1-1
AFW Pp 1-2 1-3
Close
MU-0-284
Close
MU-0-286
STR 97
STR 98
To Fire Pumps Suction
From Raw Water Reservoir

APPENDIX D

RHR TRAIN/COMPONENT FAILURE

A. FAILURE IN RHR TRAIN NO. 2

ACTIONS

1. If RHR pump No. 2 failed, containment sump to RHR valve 8982B or RHR pump No. 2 to SI suction valve 8804B failed to open proceed with steps a. to e. to provide charging and SI pump suction from RHR pump No. 1.

   a. Verify RHR pump No. 1 crosstie valves to SI pump suction 8807A or B open.

   b. Close or verify closed Train No. 2 to SI pump suction valve 8804B.

   c. Throttle HCV-638 to ensure adequate suction for the SI and centrifugal charging pumps (approximately 20 psig above containment pressure) while maintaining RHR pump motor current less than 57.5 amps.

   d. Return to the OP-2 procedure step that was left to perform Appendix A.

COMMENTS

1. Monitor RWST level, when the level reaches the lo lo alarm setpoint, trip all safeguards pumps taking suction from the tank.

APPENDIX D

B. FAILURE IN RHR TRAIN NO. 1

ACTIONS

1. If RHR pump No. 1 failed, containment sump to RHR valve 8982A or RHR pump No. 1 to charging pumps suction valve 8804A failed to open, proceed with steps a. to e. to provide charging and SI pump suction from RHR pump No. 2.

   a. Verify RHR pump No. 2 crosstie valves to SI pump suction 8807A and B open.

   b. Close or verify closed Train No. 1 to SI pump suction valve 8804A.

   c. Throttle HCV-637 to ensure adequate suction for the SI and centrifugal charging pumps (approximately 20 psig above containment pressure) while maintaining RHR pump motor current less than 57.5 amps.

   d. Return to the OP-2 procedure step that was left to perform Appendix A.

COMMENTS

1. Monitor RWST level, when the level reaches the lo lo alarm setpoint, trip all safeguards pumps taking suction from the tank.

APPENDIX E

1. If the vital busses lose voltage prior to resetting the safety injection signal, the vital busses will automatically load the vital equipment given below. Verify the equipment has been loaded by observing breaker lights on the control board.

2. If the vital busses lose voltage after the safety injection signal has been reset, load or verify loaded the equipment given below onto the vital busses manually. Allow approximately 4 seconds between loading of each piece of equipment onto a given vital bus. Load or verify that the CFCU are running in low speed.

VITAL BUS F	VITAL BUS G	VITAL BUS H
D/G 1-3	D/G 1-2	D/G 1-1
MCC 1-F	MCC 1-G	MCC 1-H
CC Pp 1-1	CC Pp 1-2	SI Pp 1-2
SI Pp 1-1		
CFCU 1-2	RHR Pp 1-1	RHR Pp 1-2
CFCU 1-1	CFCU 1-3	CFCU 1-4
CCW Pp 1-1	CFCU 1-5	CCW Pp 1-3
ASW Pp 1-1	CCW Pp 1-2	AFW Pp 1-2
AFW Pp 1-1	ASW Pp 1-2	

3. Load the Containment Spray Pumps only if they were running prior to the blackout.

VITAL BUS G	VITAL BUS H
Cont Spray Pp 1-1	Cont Spray Pp 1-2

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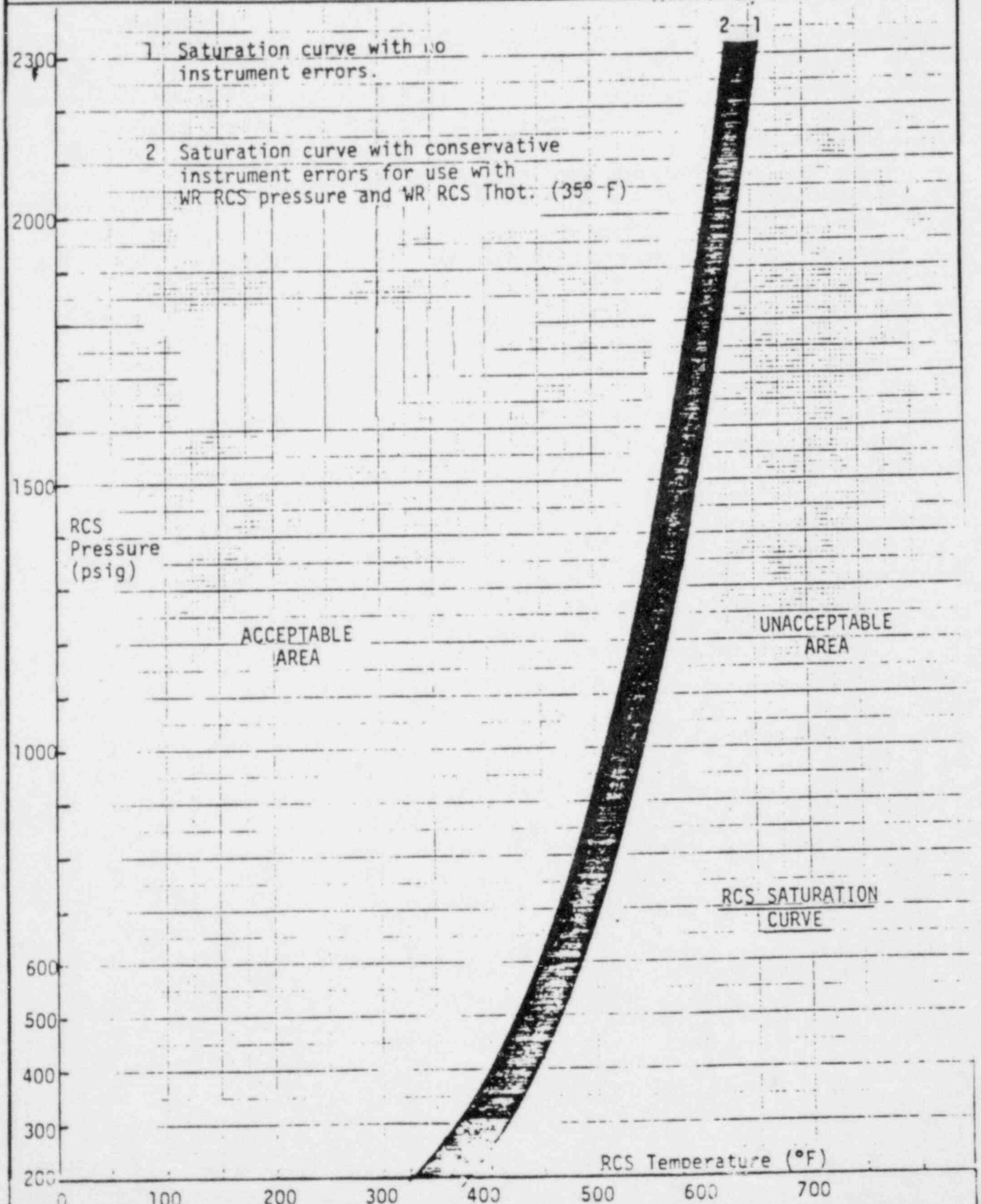
TITLE: LOSS OF SECONDARY COOLANT



APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Designate this event a Notification of Unusual Event. Notify plant staff and organizations required for this classification by Emergency Procedure G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

   b. If the steam line break is accompanied with known primary to secondary leakage > 10 gpm as indicated by high radiation alarms on the air ejector, steam generator blowdown, sample or by an increase in steam generator water level in an unexpected manner, designate this event as an Alert. Notify plant staff and response organizations required by EP G-2 and EP G-3 in accordance with EP G-1.

   c. If the primary to secondary leakage is estimated at > 50 gpm with fuel damage evident from a gross failed fuel monitor alarm or primary sample, designate this event as a Site Area Emergency. Notify plant staff and response organizations required by EP G-2 and EP G-3 in accordance with EP G-1.

TITLE: STEAM GENERATOR TUBE RUPTURE

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the operating steps to be taken in the event of a steam generator tube rupture. It is assumed that reactor trip and safety injection actuations have occurred. The operator should have already performed Emergency Operating Procedure OP-0, "Reactor Trip with Safety Injection".

SYMPTOMS

(See OP-0 symptoms)

AUTOMATIC ACTIONS

(See OP-0 Automatic Actions)

OBJECTIVES

1. To minimize the release of radioactive material by identifying and isolating the faulted steam generator and by reducing reactor coolant system pressure below the steam generator valve setting (1065 psig).

2. To establish the capability to supply feedwater to all steam generators and to isolate feedwater to the faulted steam generator.

3. To maintain the ability to remove the necessary residual heat from the reactor through the intact steam generators via the steam dump valves to the condenser or the atmosphere.

4. To maintain the reactor coolant system in a subcooled state during the recovery.

5. To prevent overflooding of the faulty steam generator.

IMMEDIATE OPERATOR ACTIONS

(See OP-0 Immediate Operator Actions)

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Sound the site emergency alarm.	
2. Contact Chemistry & Radiation Department to sample containment atmosphere and all steam generators for abnormal radiation.	2. The steam generator samples are essential for subsequent recovery actions.
3. Identify the faulted steam generator by one of the following methods.	3. While attempting to identify and isolate the faulted steam generator, continue with this procedure up to step 17.

a. Observe steam generator water levels. The faulted steam generator should experience an unexpected rise in level. If required, momentarily reduce auxiliary feedwater flow to the steam generators and attempt to identify the faulted steam generator by level indications.

b. Reset Containment Isolation Phase A, Train A and Train B.

1) If a high blowdown radiation signal is NOT present, open or check open the inside containment SG blowdown isolation valves and open the sample valves FCV 250, 248, 246 and 244 one at a time to identify the faulted steam generator. Allow time between opening to allow sample flow to contact the radiation element. The steam generator blowdown with high radiation identifies the faulted steam generator.

ACTIONS	COMMENTS
2) If high radiation level on steam generator blowdown liquid monitor has isolated steam generator blowdown and blowdown samples, determine the faulted steam generator as follows. First, check open or open the IC blowdown isolation valves, then cut in the S/G blowdown Hi Rad switch on VB-3 to override the high radiation trip signal and open FCV's 250, 248, 246 and 244 steam generators 1 through 4 blowdown sample valves, one at a time as necessary to compare radiation levels on each. The SG with the high radiation in the sample is the faulted SG.	
3) From the steam generator samples, identify the faulted Steam Generator by observation of abnormally high radiation in any one steam generator.	
4. When the faulted steam generator has been positively identified, <u>ISOLATE THE FAULTED STEAM GENERATOR.</u>	
a. Stop all AFW flow to the <u>faulted</u> steam generator.	a. Monitor the water level in the faulted steam generator throughout this procedure.

ACTIONS		COMMENTS	
b.	Close or verify closed, the following valves associated with the faulted steam generator. (VB3). Main Steam Isolation Valve, Main Steam Isolation Valve Bypass Valve, IC and OC Blowdown Valve, Blowdown Sample Valve, 10% Atmospheric Relief Valve.	b.	The steam generator safety valves on the faulted steam generator may lift during this operation.
5.	If the main steam isolation valves or the main steam isolation valve bypass valve on the faulted steam generator will not close, then close the main steam isolation valves and bypass valves on ALL nonfaulted steam generators and verify the nonfaulted steam generators 10% atmospheric relief valves maintaining steam generator pressure approximately 1035 psig.		
6.	Verify closed and place in manual the 10% atmospheric relief valve on the faulted steam generator.		
7.	If steam generator No. 2 is the faulted steam generator, close the aux. feedwater pump steam supply FCV 37. If steam generator No. 3 is the faulted steam generator, close FCV 38.	7.	This will terminate the activity release from the faulted steam generator via the steam driven aux. feed pump.
8.	Verify all pressurizer PORV's are closed. Verify by lit position lights that power is available to the pressurizer power operated relief valve backup isolation valves.	8.	Verify by position lights and discharge line temperature indication.

ACTIONS	COMMENTS
9. Monitor the core exit thermocouple temperature for indications of inadequate core cooling. If indications of inadquate core cooling exist, perform Appendix C of this procedure.	9. Indications of inadequate core cooling are given in Appendix C of this procedure.
10. If RCS wide range pressure continues to decay below 1220 psig or is below 1220 psig and stable.	10. NOTE: The conditions for stopping RCP must be continuously monitored through step 18.
a. Again verify a minimum of one charging pump delivering flow and one SI pump delivering flow to the RCS,	
b. THEN, STOP all four reactor coolant pumps. Maintain seal water flow to the RCPseals by manually adjusting the reciprocating charging pump speed or FCV 128.	
c. Close the Centrifugal Charging pump recirculation valves (8105 and 8106)	c. NOTE: If the W.R. RCS pressure is increased above 2000 psig, reopen valves 8105 and 8106 to prevent pump damage.
d. If component cooling water to the RCP's is isolated due to a containment Phase B isolation, stop all RCP's within 5 minutes and maintain seal flow as above.	
11. If the faulted steam generator was isolated by closure of its MSIV (per step 4 of this procedure) perform steps a. & b. below. If it was isolated as per step 5, go to step 12.	11. If the faulted steam generator has been identified, do not dump steam from the faulted steam generator. Maintain containment Phase A Isolation until necessary to reset for system operations.

ACTIONS	COMMENTS
a. If the condenser is available, open or verify open the nonfaulted steam generators main steam isolation valves and establish steam dump to the condenser. Transfer the steam dump control to the pressure control mode and verify set pressure at NO load pressure.	
b. If the condenser is not available, verify the 10% atmospheric relief valves holding steam pressure below the safety valve setpoint.	
12. Verify or establish power on the 480 volt vital buses, F, G and H.	12. This power, plus the normal and backup bottled air supply, will assure power sources available for at least one PZR PORV, steam generator PORV's and charging and letdown flowpaths. If loss of offsite power occurs, the letdown path will be to the PRT via the relief valve downstream of the orifice valves.
13. Verify RCS T_{avg} is at or approaching $T_{no\ load}$ under the influence of condenser or atmospheric relief valves from the nonfaulted steam generators. Adjust steam dump if required to achieve $T_{no\ load}$.	
14. Maintain maximum AFW flow until the steam generator water levels are in the narrow range. When the SG water levels approach 33% NR, verify automatic steam generator level control.	14. Observe the water levels closely for unexplained changes in one steam generator which may identify the faulted steam generator.

ACTIONS	COMMENTS
15. Monitor the condensate storage tank and upon reaching approximately 10% level, perform a. or b. below.	15. If the CST lo lo level alarm occur the operator has approximately 25 minutes to perform Steps a. or b.
a. Verify a level in the raw water storage reservoir; then open FCV-436 and FCV-437 (Reservoir supply to AFW pumps). Allow the AFW pumps to run during the transfer. Monitor the AFW flow closely. If AFW flow is lost, trip all 3 AFW pumps until the transfer is complete, then restart the pumps.	
b. If the raw water storage reservoir is not available, go to Appendix A (AFW Pump Suction Supply from Fire Water Tank Procedure). Allow the AFW pumps to run during the transfer. Monitor the AFW flow closely. If AFW flow is lost, trip all 3 AFW pumps until the transfer is complete, then restart the pumps.	
16. If the RCS pressure is above the shutoff head of the RHR pumps, RESET SAFETY INJECTION and stop both RHR pumps.	16. CAUTION: 1) If the RCS pressure falls below the shutoff head of the RHR pumps, restart the pumps to deliver water to the RCS CAUTION: 2) Automatic reinitiation of safety injection will not occur after this step since the reactor trip breakers are open. If the operator has indication that an SI is required after this step, he must initiate it manually.

ACTIONS	COMMENTS
	16. (Cont)
	CAUTION:3)If loss of offsite power occurs after resetting safety injection, it will be necessary to load the safeguards equipment onto the vital buses manually. If safety injection is reinitiated manually after the loss of offsite power, the vital buses will automatically sequentially load the safeguard equipment. If loss of offsite power occurs, go to Appendix B (Blackout With SI Emerg. Loading of Vital Buses).
17. DO NOT PROCEED BEYOND THIS STEP UNTIL THE FAULTED STEAM GENERATOR IS IDENTIFIED AND ISOLATED.	
18. Begin a Rapid cooldown of the RCS to 500 degrees F using only the nonfaulted steam generators.	
a. If the faulted steam generator was isolated by closure of the main steam isolation valves associated with the nonfaulted steam generators, dump steam only from the nonfaulted steam generators through the 10% atmospheric relief valves. If isolated per step 4, use step b. or c. below.	
b. Use the condenser steam dumps if the condenser is available. Go to steam pressure control mode if not already in this mode. Place the steam pressure controller in MANUAL and increase the demand as necessary	b. This is the preferred method. When P-12 is reached, select Bypass Interlock on the Steam Dump Interlock Selector switches to permit the cooldown.

ACTIONS	COMMENTS
c. If the condenser is not available, use the 10% atmospheric relief valves from the nonfaulted steam generators.	
19. Continue to monitor containment conditions, if containment sump level rises abnormally or if a containment sample (if available) indicates high activity in containment, go to OP-1 for further accident recovery.	
20. After the RCS has been cooled to 500 degrees F, if required, depressurize the RCS to the value equal to the faulted steam generator pressure. If pressurizer level has been off scale low, the level will probably return during this operation. Maintain minimum 35 degrees F subcooling during this operation.	20. During subsequent controlled RCS depressurization, the criteria for tripping RCP's on low pressure no longer applies.
a. If RCP's are in service, use pressurizer spray to reduce RCS pressure.	
b. If normal spray is not available, open one pressurizer Power Operated Relief Valve (PORV) to reduce RCS pressure. Verify closure of the valve by observing position indication and discharge line temperature decreasing, and if required, close the backup isolation valve.	b. NOTE: It may take 2-3 minutes for PORV discharge line temperature to start decreasing.

ACTIONS

21. As RCS pressurize decreases, due to PZR spray or the PZR PORV being open, monitor PZR level and stop the depressurization when:

    a. PZR level exceeds 85%,

    b. OR RCS pressure decreases to the pressure of the faulted steam generator.

    Verify closure of the PORV by position indication and discharge line temperature. Verify closure of the spray valve by position indication.

22. Continue to monitor RCS pressure and pressurizer water level.

    a. If pressurizer level continues to rise or is stable with continued RCS pressure decreasing after the depressurization is terminated, suspect leakage from the pressurizer steam space. If this condition persists and the PRT rupture disc is ruptured go to OP-1, "Loss of Coolant Accident".

    b. If the pressurizer level continues to rise with rising RCS pressure,

    AND PRT conditions are stable, the SI flow is greater than the tube rupture flow.

COMMENTS

22. a. Monitor PRT conditions and relief line temperatures to detect a possible stuck open valve. Close the appropriate PORV isolation valve if leakage is suspected.

# ACTIONS	COMMENTS
c. DO NOT continue in this procedure until the conditions of b. above are observed.	
23. When RCS pressure has increased by 200 psig after the termination of the depressurization,	
a. AND Pressurizer water level is greater than 22%	
b. AND indicated subcooling from the nonfaulted steam generator loops is greater than 35 degrees F	
c. THEN allow one centrifugal charging pump to operate for normal charging and RCP seals and shutdown the remaining charging, and SI pumps while maintaining operable safety injection flowpaths.	When the charging and SI pumps are shutdown the RCS pressure should decrease to the value of the faulted steam generator.
24. Establish normal charging.	24. Continue monitoring PZR level. If PZR level cannot be maintained above 22%, or indicated subcooling from the nonfaulted steam generator loops cannot be maintained greater than 35 degrees F, manually reinitiate SI and return to step 20 and continue.
a. Reset containment Phase A isolation if required.	
b. Check open or open normal charging valve 8146.	
c. Check closed or close charging to aux. spray valve 8145 and alt. spray bypass valve 8148 and alternate charging valve 8147.	
d. Open charging line isolation valves 8107 and 8108.	
e. Close the BIT inlet and outlet valves 8803A and B and 8801A and B.	

ACTIONS COMMENTS

f. Adjust HCV-142 and FCV-128 or reciprocal charging pump speed to achieve RCP seal flow and charging flow as required to maintain pressurizer level greater than 22%.

g. Open RCP seal return valves, 8100 and 8112. Check seal flow normal.

h. Open RCP No. 1 seal bypass valve (8142) if RCS pressure is less than 1500 psig.

25. Establish normal letdown.

a. Check open or open letdown valves LCV-459 and 460.

b. Open letdown isolation valve 8152.

c. Open one 75 gpm letdown orifice valve.

d. Verify PCV-135 opening by observing letdown flow.

26. Establish VCT makeup and transfer charging pumps suction to VCT.

a. Adjust VCT makeup blend to the cold Xe free concentration.

b. Open VCT outlet valves LCV-112B&C.

c. Close RWST to charging pump suction valves 8805 A&B.

d. Verify divert valve LCV-112A in AUTO.

27. Reestablish the use of the pressurizer heaters to control pressure. If required, transfer the pressurizer backup heaters groups 2 and 3 to the vital 480 volt buses G&H.

ACTIONS		COMMENTS	
28.	If possible, establish conditions for starting No. 1 and No. 2 RCP's if the pumps have been shutdown.	28.	The low pressure pump trip criteria no longer applies.
	a. When conditions are established, start both RCP's.	a.	Start both pumps to deliver spray regardless of whether these pumps are associated with a faulted steam generator.
29.	If No. 1 and/or No. 2 RCP is running, shutdown No. 3 and and No. 4 RCP's.	29.	This step will reduce the primary heat load.
30.	With the plant in a stabilized condition, determine if the condenser will be available to receive steam dump during the subsequent cooldown.		
	a. If the condenser is available, perform steps 31, 32 and 33 SIMULTANEOUSLY.	a.	Failure to perform steps 31, 32 and 33 simultaneously may result in a loss of PZR level control.
	b. If the condenser is not available, notify the Chemistry and Radiation Protection Department that a controlled activity release will be occuring. If time permits, verify that all faulted steam generator samples have been taken, then perform steps 31, 32, and 33 SIMULTANEOUSLY.	b.	Consult the Vol. 9 curves to determine how long the plant can remain at hot standby before proceeding to cold shutdown.
31.	Begin a controlled cooldown of the RCS using steam dump from the <u>NONFAULTED</u> steam generators only. Verify as steam dump begins that AFW system is maintaining ALL steam generator levels in automatic.		

ACTIONS	COMMENTS
Place the LCV for the faulted steam generator in automatic and verify normal AFW response to the faulted steam generator existing level. If the faulted steam generator level is low, verify feedwater flow to the steam generator; if level is high, NO feedwater flow to the steam generator.	
a. Maintain an RCS cooldown rate of about 50 degrees F/HR.	
b. Use the condenser dump valves by reducing the pressure setpoint. If the condenser is not available, use the 10% atmospheric relief valves. If the faulted steam generator was isolated as per step 5 of this procedure, use the 10% atmospheric relief valves.	
32. Simultaneous with the cooldown, dump steam from the faulted steam generator to the condenser.	
a. Bleed steam to the condenser using the Main Steam Isolation Valve BYPASS VALVE.	a. THIS IS THE PREFERRED METHOD.
b. If the condenser is not available, use the 10% atmospheric steam relief valves.	b. THIS IS NOT THE PREFERRED METHOD.
33. Simultaneous with the faulted steam generator pressure decay, control RCS pressure approximately the same as the faulted steam generator pressure.	33. This will minimize the mass RCS and the faulted steam generator. Stay within the limits on the Tech. Spec. cooldown curves during this operation.
a. Use PZR heaters and one of the following:	
b. Use normal PZR spray if possible OR	

ACTIONS	COMMENTS
c. Use auxiliary spray if letdown is in service OR	
d. Use PZR PORV intermittently if required. If the PZR PORV is used, continuously monitor PRT pressure, temperature, and level and take appropriate actions to maintain PRT integrity. Verify PORV closure by position indication and PRT conditons. If a RCS leak to the PRT is identified, close the PORV isolation valve.	
34. Determine the cold Xe Free Shutdown margin and borate if required to that concentration.	
35. At approximately 800 psig RCS pressure close all 4 accumulator injection isolation valves.	
36. When the RCS hot leg temperatures are reduced to less than 350 degrees F and RCS pressure is less than 400 psig, place the RHR in service using Operating Procedure B-2. (Residual Heat Removal System).	36. Do not collapse the PZR bubble.
37. Continue the cooldown in this mode until the RCP is stopped, then continue to control RCS and faulted steam generator pressures until the RCS hot leg temperatures are below 200 degrees F, then use auxiliary spray until the RCS pressure and faulted steam generator pressures equilibrate.	37. Use Operating Procedure L-5 (Plant Cooldown From Minimum Load to Cold Shutdown) in conjunction with this procedure during the cooldown. Enter L-5 at the point where RHR is to be put into service.
38. Continue operation of the RHR, letdown and charging as required.	

APPENDIX A

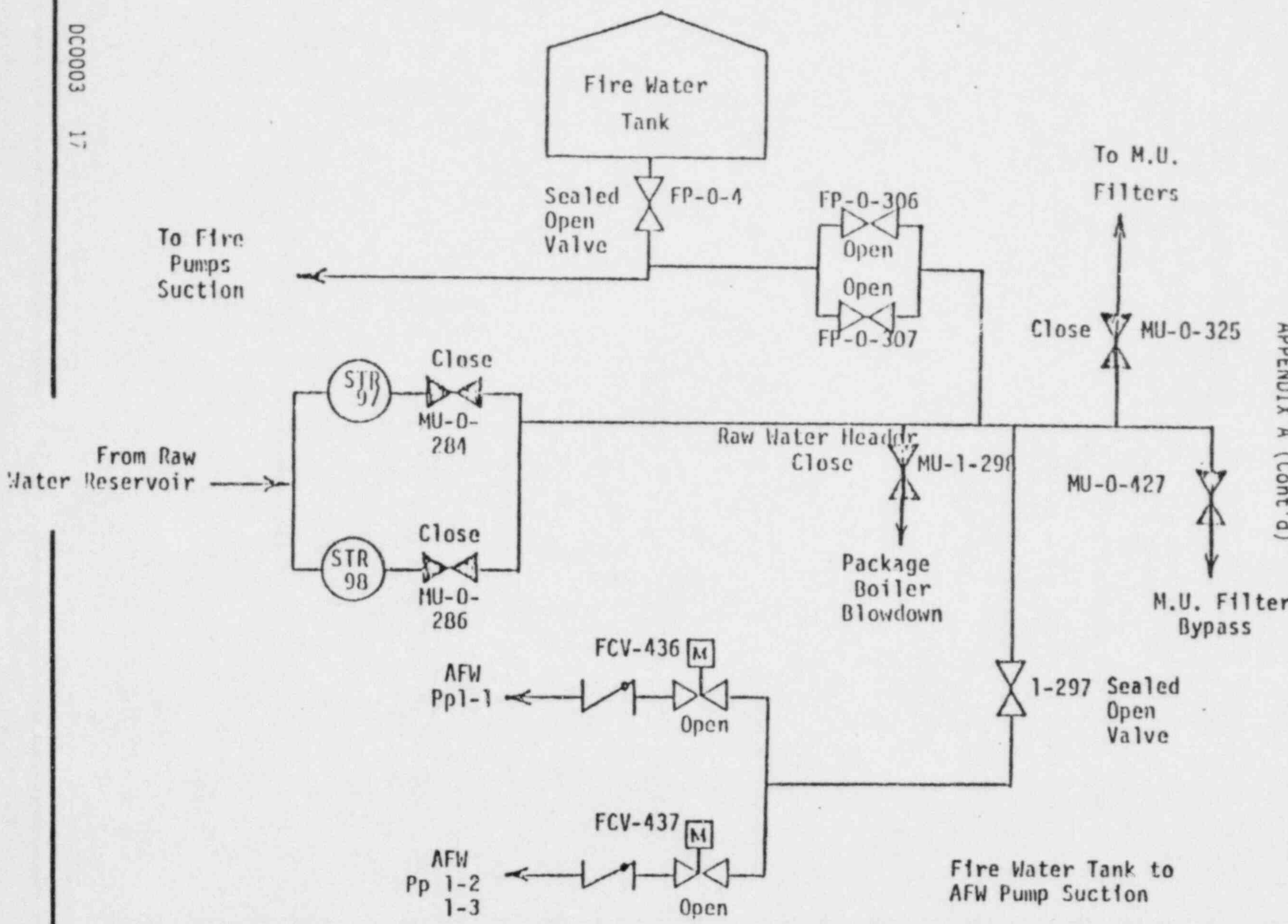
AUXILIARY FEED PUMP SUCTION SUPPLY FROM FIRE WATER TANK

The operator has 20 minutes to perform this operation after the lo lo level alarm on the condensate storage tank and before the AFW pumps lose suction. This provides sufficient time; however, the operator must not delay and must carry out the valve line up in order as written.

If the AFW pumps are being supplied from the raw water reservoir and a seismic event occurs with resultant loss of AFW suction and auxiliary feedwater flow to the steam generators, the steam generators will boil dry in about 30 minutes. Under these conditions, it is especially important to expedite this procedure and reestablish AFW flow to the steam generators prior to the reactor losing its heat sink.

ACTIONS	COMMENTS
Using the attached drawing, proceed to supply the AFW pumps suction from the fire water tank.	
1. Close or verify closed MU-0-284 and MU-0-286.	1. Closing these valves prevents losing fire water out a possible break in the reservoir supply line.
2. Close or check closed MU-1-298.	
3. Close or check closed MU-0-325.	
4. Close or check closed MU-0-427.	
5. Open FP-0-306 and FP-0-307.	
6. Notify the control room that the suction for the AFW pumps is now available from the fire water tank.	
7. From the control room open FCV-436 and 437.	
8. Proceed to the auxiliary feedwater pumps and vent the pump casings if required to remove air.	

APPENDIX A (Cont'd)



APPENDIX B

BLACKOUT WITH SAFETY INJECTION EMERGENCY LOADING OF VITAL BUSES

1. If the vital buses lose voltage prior to resetting the safety injection signal, the vital buses will automatically load the vital equipment given below. Verify the equipment has been loaded by observing breaker lights on the control board.

2. If the vital buses lose voltage after the safety injection signal has been reset, load or verify loaded the equipment given below onto the vital buses manually. Allow approximately 4 seconds between loading of each piece of equipment onto a given vital bus. Load or verify that the CFCU's are running in Low Speed.

VITAL BUS F	VITAL BUS G	VITAL BUS H
D/G 1-3	D/G 1-2	D/G 1-1
MCC 1-F	MCC 1-G	MCC 1-H
CC Pp 1-1	CC Pp 1-2	SI Pp 1-2
SI Pp 1-1		
CFCU 1-2	RHR Pp 1-1	RHR Pp 1-2
CFCU 1-1	CFCU 1-3	CFCU 1-4
CCW Pp 1-1	CFCU 1-5	CCW Pp 1-3
ASW Pp 1-1	CCW Pp 1-2	AFW Pp 1-2
AFW Pp 1-1	ASW Pp 1-2	

3. Load the containment spray Pumps only if they were running prior to the blackout.

VITAL BUS G	VITAL BUS H
Cont Spray Pp 1-1	Cont Spray Pp 1-2

APPENDIX C

DETERMINATION OF ADEQUATE CORE COOLING

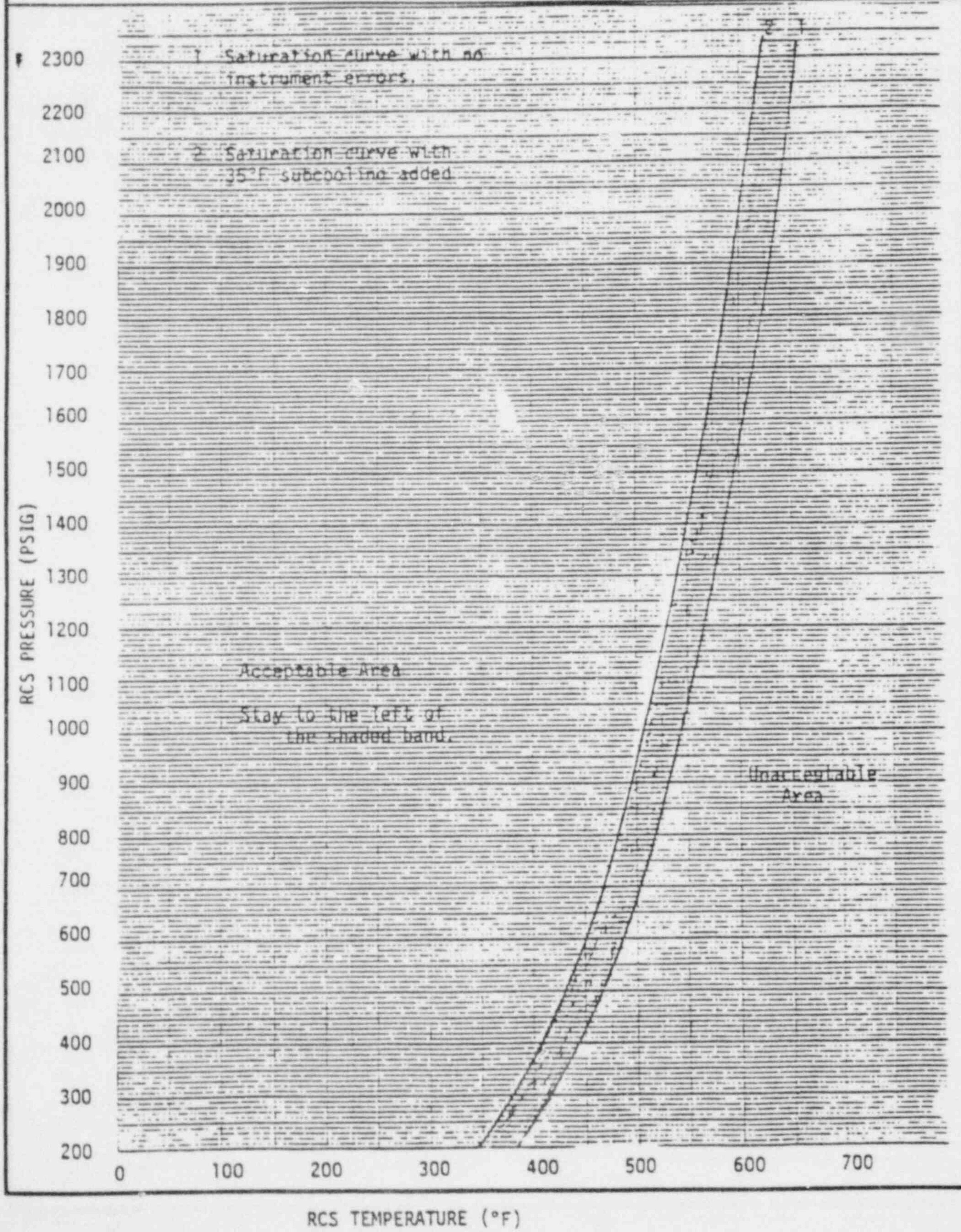
This appendix provides the guidance to determine adequate core cooling if inadequate core cooling is suspected. Further, the instructions for regaining adequate core cooling is presented.

ACTION	COMMENTS
1. Monitor the core exit thermocouple temperatures. a. If the P-250 is available go to step 2. b. If the P-250 is <u>not</u> available go to step 3.	
2. If 5 or more P-250 thermocouple readings exceed 1200 degrees F, notify the Shift Foreman that inadquate core cooling exists and go to step 5. If there are <u>not</u> 5 or more that exceed 1200 degrees F, discontinue this appendix but continue to monitor the thermocouple readings.	
3. At the thermocouple incore board in the control room, Monitor 10 core-centered thermocouples. If any 3 of the 10 thermocouples exceed 700 degrees F (pegged hi) proceed to step 4. If there are <u>not</u> 3 thermocouples that exceed 700 degrees F, discontinue this appendix but continue to monitor the thermocouple readings.	
4. If SI flow to the RCS and AFW flow to the steam generators cannot be verified, notify the Shift Foreman that inadequate core cooling may exist	

ACTION	COMMENTS
and go to step 5. If both SI flow to the RCS and AFW flow to the steam generators can be verified, discontinue this Appendix but continue to monitor the thermocouple readings.	
5. The Shift Foreman will verify if inadequate core cooling exists using the appropriate steps above. If inadequate core cooling exists the Shift Foreman will direct operations as follows:	
a. Declare a General Emergency. Implement the instructions given in Emergency Procedure G-1 regarding on and offsite protective actions.	
b. Attempt to establish SI flow to the RCS and AFW flow to the steam generators.	
c. Continue monitoring core outlet temperature to determine the effectiveness of the remaining actions.	
d. DEPRESSURIZE THE RCS by method 1 or 2 below.	
1) Dump steam to the condenser or atmosphere if the steam generator levels are in the narrow range and AFW flow is evident.	1) THIS IS THE PREFERRED METHOD.
2) Verify the SIS or charging pumps are running and available to deliver water to the RCS THEN Open the pressurizer PORV's.	2) Opening the PORV's will provide a drop in RCS pressure sufficient to allow the SI flow required to cool the core. This method is to be used only if 1) (above) is ineffective.

ACTIONS	COMMENTS
e. If no means of depressurization are available, or if the depressurization did not result in decreasing core exit thermocouple temperatures, THEN START one RCP if possible. If the RCP fails after starting, replace the lost RCP with any remaining RCP.	e. Attempt to establish CCW and seal water flow to the pump; however, if CCW and/or seal water flow cannot be established, proceed to start a RCP. The pump must be started to move coolant through the core.

TITLE: STEAM GENERATOR TUBE RUPTURE



APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION

1. When this emergency procedure has been activated and upon direction from the Shift Foreman proceed as follows:

   a. Designate this event an Alert. Notify plant staff and response organizations required for this classification by Emergency Procedure G-2 "Establishment of On-Site Organization" and Emergency Procedure G-3 "Notification of Off-Site Organization" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

   b. Designate this event a Site Area Emergency if steam generator tube leakage coincides with a loss of off-site power indicated by Appendix B of this procedure and inability to restore power to the non-vital 12KV and 4K busses (D and E). Notify plant staff and response organizations required by EP G-2 and EP G-3 in accordance with EP G-1.

   c. In the event inadequate core cooling is verified per Appendix C, reclassify this event as a General Emergency. Notify plant staff and response organizations required by EP G-2 and EP G-3 in accordance with EP G-1.

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER	EP OP-5
REVISION	5
DATE	2/11/82
PAGE	1 OF 6

TITLE: EMERGENCY OPERATING PROCEDURE
REACTOR TRIP WITHOUT SAFETY INJECTION

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the steps to be taken following a reactor trip. A reactor trip does not normally result in an emergency situation; however, it may, and therefore is covered in an emergency procedure. In addition, many true emergencies will produce a reactor trip and this procedure should be followed in those cases. The possibility of a ATWS should always be considered.

SYMPTOMS

1. Reduction of loss of forced reactor coolant flow (2 RCP's NOT running above P-7, 1 RCP NOT running above P-8) due to pump or motor failures, undervoltage or underfrequency on the reactor coolant pump busses, or pump motor breaker trip.

2. Low pressurizer pressure, when the plant is being operated at or above P-7.

3. Power range high neutron flux.

4. High neutron flux rate (positive or negative).

5. Intermediate range high neutron flux, when not manually bypassed (p-10).

6. Source range high neutron flux, when not manually bypassed (P-6).

7. Overpower ΔT.

8. Overtemperature ΔT.

9. Turbine trip, when the plant is operated above P-7.

10. High Pressurizer pressure.

11. High pressurizer water level, when the plant is operated above P-7.

12. Low-low steam generator water level.

13. Low feed flow/high steam flow and low water level.

14. Simultaneous General Warning Alarm in both trains of the Plant Protection system.

15. Seismic Trip

16. Manual Reactor Trip

AUTOMATIC ACTIONS

1. Reactor Trip
2. Turbine trip
3. Steam dump activation
4. Generator trip and transfer of electrical power to startup transformers

OBJECTIVES

1. To ensure the reactor and turbine are tripped
2. To ensure a heat sink for the reactor
3. To stabilize the plant in a subcritical condition

IMMEDIATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Check that the reactor has tripped and manually trip it if it has not.	1. Verify reactor trip by noting:
a. Use the manual reactor trip switch.	a. All DPRI rod bottom lights on.
b. If all else fails, deenergize 480V load centers 13D and E by opening breakers 52-HD-13 and 52-HE-4.	b. Nuclear power decreasing.
	c. A negative startup rate is observed on the intermediate range and/or source range rate meters.
2. Emergency Borate 100 ppm for each control rod not fully inserted. Use OP-6.	
3. Check that the turbine has tripped and manually trip it if necessary.	3. Verify turbine trips by noting:
a. Use the manual turbine trip switch.	a. Stop valves closed (position lamps).
b. If necessar use the trip level on the gove...or pedestal.	NOTE: Generator trip is delayed ≈30 seconds after turbine trip.

ACTIONS	COMMENTS
4. Verify RCS heat removal by checking Tavg is approaching or at no-load conditions.	
5. Start or check running both motor driven auxiliary feedwater pumps, check flow to all steam generators.	5. If motor driven pumps fail to start, start or check running the turbine driven auxiliary feed pump.

SUBSEQUENT OPERATOR ACTIONS

ACTIONS

COMMENTS

1. Transfer the steam dump system to the steam pressure mode with a setpoint of 1005 psig.

2. Check that the feedwater control valves close when Tavg decreases below 554°F. Close them manually if necessary.

3. Check that all 4 and 12KV busses transfer to start-up power. If off-site power is lost, go to OP-4.

4. Monitor reactor coolant system parameters to determine if they are returning to their expected values following the initial transient response to the reactor trip.

   a. Reactor coolant average temperature should approach 547°F.

   b. Pressurizer level should approach 22%.

   c. Pressure should remain above 1950 psig during and immediately following the transient then return to approximately 2235 psig.

ACTIONS

d. High pressurizer pressure will open the pressurizer Power Operated Relief Valves. If these valves open during the transient, verify their subsequent closure following the reduction of primary pressure by monitoring valve position indications, relief valve discharge header temperature trends, and pressurizer relief tank temperature.

5. Transfer the NIS recorder to monitor one IR and one SR channel.

6. Check the turbine-generator coasting down properly.

   a. All turbine drain valves open.

   b. The AC bearing oil pump and the high pressure seal oil backup pump start automatically.

   c. The lift pump starts at about 600 RPM.

   d. The turning gear engages automatically at or near zero speed.

7. Shutdown the main feedwater pumps.

8. Maintain condenser vacuum; if vacuum is lost use the 10% atmospheric dump valves to control steam generator pressure.

9. Establish and maintain hot standby operation, verify shutdown margin per STP R-19, and adjust RCS boron concentration if necessary.

10. If condenser vacuum is lost, check the level in the condensate storage tank to determine how long the unit can be maintained in hot standby prior to going to cold shutdown. Refer to Curves presented in Vol. 9 of the plant manual.

COMMENTS

d. Should a pressurizer Power Operated Relief Valve fail to reseat after pressurizer pressure has dropped below the nominal reset value, close the associated backup isolation valve for the malfunctioning Power Operated Relief Valve.

ACTIONS

COMMENTS

11. When the cause of the reactor trip has been determined and corrected, the reactor may be returned to operation in accordance with Administrative Procedure A-100.

12. If conditions permit, open the generator motor-operated disconnect and align the auxiliary power system to backfeed from the 500KV switchyard to feed the plant 4KV and 12KV busses.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this a significant event and within one hour notify the NRC Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to operating procedure O-4 "Operating Order (1 hour Reporting Requirement to NRC)" for guidance on the format for the report. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Event).

   c. When the cause of the reactor trip has been determined, this and the subsequent plant conditions should be reviewed against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
LOSS OF CONDENSER VACUUM

APPROVED: [signature] 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the steps to be taken following a complete loss of condenser vacuum which results in a unit trip. A low vacuum trip results in blocking the 40% condenser dump valves and the 35% atmospheric dump valves leaving only the 10% pressure relief valves available. These valves have sufficient heat removal capacity to prevent actuation of the pressurizer relief or safety valves and probably prevent actuation of the steam generator safety valves.

Reduced condenser vacuum (above the low vacuum trip point) is covered in Operating Procedure C-6.

SYMPTOMS

1. Turbine Trip alarm - window PK12-11.
2. Feedwater pump trip alarm - window PK09-12.
3. Reactor Trip Actuated alarm - window PK04-14
4. Condenser Interlock C-9 actuated - window PK08-14

AUTOMATIC ACTION

1. Turbine trip on low vacuum
2. Generator trip
3. Reactor trip
4. Block of 40% condenser steam dump
5. Block of 35% atmospheric steam dump
6. Trip of both main feedwater pumps
7. Automatic transfer of all electrical power supplies to the startup source
8. Start of both motor driven auxiliary feedwater pumps
9. Activate steam generator 10% atmospheric relief valves
10. Possible activation of steam generator safety valves

OBJECTIVES

1. Verify the proper actuation of all systems required to respond and maintain the reactor in Hot Standby using the 10% atmospheric relief valves.
2. Protect the main turbine-generator and main feedwater pumps by placing them on turning gear.
3. Investigate and correct the cause of the loss of condenser vacuum to allow the unit to return to service.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Verify reactor trip and follow instructions of reactor trip procedure.	1. "Reactor Trip Without Safety Injection" Operating Procedure OP-5.
2. Verify that all automatic actions listed above took place. Actuate them manually if necessary.	

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. Control auxiliary feedwater flow to each steam generator to prevent excessive cooldown and/or water hammer.	
2. Maintain the reactor in the Hot Standby condition.	
3. Place the main turbine-generator on turning gear.	
4. Place both main feedwater pumps on turning gear.	
5. Investigate and correct the cause of loss of vacuum.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

    a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

    b. Designate this a significant event and within one hour notify the NRC Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for guidance on the format for the report. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

    c. When the cause of the loss of condenser vacuum has been determined, this and the subsequent plant conditions should be reviewed against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-8
REVISION 5
DATE 2/11/82

TITLE: EMERGENCY OPERATING PROCEDURE
CONTROL ROOM INACCESSIBILITY

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

These instructions are provided to cover those conditions prevailing when operation from the main Control Room is no longer possible due to fire, smoke, heat, chlorine, high radioactivity or other occurrences which make the Control Room uninhabitable.

SYMPTOMS

The following alarms may come in:

1. Control Room Ventilation alarm (PK5-06) due to hi radiation or hi chlorine.
2. Fire/Smoke detector alarm (PK 10-10).
3. High radiation alarms.

OBJECTIVES

1. Establish stable hot standby conditions from the hot shutdown panel.
2. To provide instructions to allow the plant to be taken to cold shutdown condition from hot standby from outside the Control Room if conditions require this.

AUTOMATIC ACTIONS

1. Possible transfer of Control Room ventilation to Mode 3 (high Chlorine) or Mode 4 (Pressurization).
2. Possible start of the diesel generators (if off-site power is lost).

IMMEDIATE OPERATOR ACTIONS

1. Trip the reactor and main unit turbine.
2. Verify reactor tripped (all rods on botton, DRPI - reactor power decreasing, Nuclear Instruments).
3. Verify turbine tripped (all four SV closed on EH panel).
4. Proceed to the hot shutdown panel.

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
The below listed equipment should have its control switch placed in the following position prior to transferring to local control to insure a bumpless transfer. a. PZR HTR GRPS 12 and 14 to the neutral position. b. The appropriate orifice isolation valve (8149 A, B,C) to the open position, and orifices not in service in the closed position. c. BA XFR PPS 11 and 12 to low speed.	Transfer of equipment to local control may remove it from various auto control schemes (i.e., press., level, etc.) Auto starts due to bus transfer or SI will still occur. Transferring BA pps & Pzr groups to local control removes them from their auto control schemes. (It may be advisable to leave them in REMOTE at this time).
1. TO MAINTAIN THE PLANT IN HOT STANDBY A. With off-site power available 1. The following equipment can be transferred to local control immediately: *a. BA XFER pump 11	
*b. Containment fan coolers 11, 13 and 14. *c. 8149A&C (letdown orifice valves).	It is advisable to leave the following equipment in AUTO (Remote) if they are controlling properly in the AUTO position, unless manual control is desired. HCV 142 FCV 128 PCV 19,20,21 and 22 (10% steam dumps) LCV 110,111,115 and 113 (AFW level control valves)

*These items must have their control transfer relays hand reset when transferring control back to Control Room (in addition to being in remote position on transfer switch).

ACTIONS | COMMENTS

d. HCV 142
e. FCV 128
f. 10% steam dump valves
g. LCV 110,111,115 and 113
h. CCW pump 11 and 13
i. ASW pump 11
j. Cent. Chg. pump 11
*k. Pzr. Heater group 14
l. AFW pump 13
*m. LCV 108
*n. LCV 109

2. The following equipment must have their control transfer cutout switch cut in prior to transferring to local control:

   a. Located inside the 480V Bus F AUX relay panel:

      *1. Containment fan cooler 12

      *2. 8149B (letdown orifice valve)

   b. Located inside the 480V Bus G AUX relay panel:

      *1. FCV 95 (AFW pump 11)

      *2. MOV 8104 (Emerg. Borate Valve)

      *3. LCV 106

      *4. LCV 107

      *5. BA XFER pump 12

      *6. CONTAINMENT fan cooler 15

*These items must have their control transfer relays hand reset when transferring control back to Control Room (in addition to being in remote position on transfer switch).

ACTIONS	COMMENTS
c. Located at 4 KV Bus G	
1. CCW pump 12	
2. ASW pump 12	
3. Cent. Chg. pump 12	
d. Located at 4 KV Bus H	
AFW pump 12	
e. LOCATED INSIDE 480V Bus 13D cutout switch panel	
*PZR heater group 12	
3. Trip the main feedwater pump turbines and ensure that they go on turning gear once zero speed is reached.	
4. Ensure that the bearing oil pump, HP seal oil backup pump and the bearing lift oil pumps are running; and that the main unit turbine goes on turning gear once zero speed is reached.	
5. Maintain the reactor at hot standby condition by:	
a. Controlling steam generators at 33% with the auxiliary feedwater controls.	a. See attached graph for determining actual VS. indic. level (approx. 63% indic. on wide range equals 33% actual on narrow range at 1005#).

*These items must have their control transfer relays hand reset when transferring control back to Control Room (in addition to being in remote position on transfer switch).

ACTIONS	COMMENTS
b. Controlling pressurizer level at 22% by use of the charging pumps and letdown valves. Do not allow level to go below 20% to ensure against letdown isolation. If letdown isolation does occur, restore letdown per Appendix A.	
c. Control pressurizer pressure at 2235 psi by use of the backup heaters as needed.	
d. Verify steam generator pressure is being maintained at approximately 1005 psi by use of the condenser steam dump. This will be an automatic function unless vacuum is lost on the condenser. Manual operation of the 10% steam dumps will then be required.	
e. Calculate the SDM for hot Xe free conditions.	
f. Stop any dilution in progress.	f. This may require local valve manipulations.
6. Establish or verify communications between the hot shutdown panel and:	6. Establish as personnel availability allows.
a. Outside telephone exchange (send an operator to the plant office).	
b. Dedicated shutdown panel	
c. 480V vital switchgear area	
d. 4 KV vital switchgear area	

ACTIONS	COMMENTS
7. Maintain shutdown margin using method a. or b. below.	
a. By opening the emergency borate valve 8104 from the hot shutdown panel.	a. Observe boric acid flow on FI at hot shutdown panel.
b. Boration may be accomplished by opening manual valve 8471 so that boric acid flows through normally open FCV-110A, but bypasses the blender, and running the boric acid pump in fast speed.	b. Indication of boric acid flow using this method will be at local FI (PM 96 on 100' El. Aux. Bldg.)
8. Place diesel generator control selector switch on excitation cubicle to LOCAL position. Verify the AUTO/TEST selector switch on local panel in the AUTO position.	8. Done locally in each D.G. room.
9. Place SU FDR breaker transfer switches to LOCAL position on cubicles.	
B. With loss of off-site power:	
1. The same as A. above except that diesels start automatically and assume the vital bus loads.	
a. Verify that the diesels have started and have assumed their normal loads. Verify 4 KV Bus F, G and H volts at the hot shutdown panel.	a. Shutdown unnecessary equip. not needed for current plant status.
b. Verify that equipment with controls located at hot shutdown panel have restarted after transfer of power to diesels. Restart locally if necessary.	

ACTIONS	COMMENTS
c. Transfer PZR heater groups 12 & 13 to emergency supply. 480V vital power breakers 52-1G-72, 52-1H-74.	c. Htr Grps 12&13 on back-up supply can ONLY be controlled (from outside the control room) by locally closing or opening their breakers for pressure control.
C. Maintain hot standby until Control Room access is restored and control has been returned to the Control Room.	C. If Control Room access is not restored rapidly, a decision must be reached on how long to remain in hot standby and when to start cooling down to cold shutdown. The basis for this decision is the amount of water inventory in the condensate storage tank. Refer to condensate curves in this procedure for guidance in making this decision.
D. Notify plant management.	
E. Declare this event an alert. Carry out the instructions given in General Appendix 2 to the Emergency Procedures (Notification of Offsite Agencies in the Event of an Emergency).	

TO TAKE THE PLANT FROM HOT STANDBY TO COLD SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

ACTIONS	COMMENTS
1. Calculate the SDM for cold Xe free conditions and borate as necessary. Energize pressurizer heaters during boration to equalize the boron concentrations in the PZR with the RCS. Alternatively, see Appendix B for spray actuation.	1. Containment entry will be required to accomplish various evolutions in this procedure, therefore, preparations should be made as soon as practicable to allow for this. Also, I&C Dept. personnel assistance will be required to execute certain steps.

ACTIONS	COMMENTS
2. Sample the RCS and PZR to ensure boron concentrations are equalized and adequate for cold shutdown.	
3. Maintain RCS temperature and press. within the operating band of the attached curves during cooldown.	3. Do not exceed 100°F/HR cooldown rate. A 50°F/HR cooldown rate is recommended.
4. Maintain cold Xe free boron conc. by closely monitoring RCS boron conc.	4. Remote readout of the boron analyzer is available at the secondary sample panel (85' El.).
5. De-energize all PZR heaters. Open and rack out breakers for Prop. heaters (52-13D-5) and group 13 heaters (52-13E-2). Place heater groups 12 and 14 in the OFF position.	5. If heater groups 12 and 13 are on backup supply, open their bkrs.
6. Commence RCS cooldown using 10% atmospheric steam relief valves and reducing pressure using pressurizer spray per Appendix B attached. (If it is desired to secure normal charging to RCS valve 8146, a pneumatic jumper will have to be installed around SV 200 in PM 90 in Containment.)	6. As an alternate means of cooldown (if condenser available) use handjack on 40% steam dump valve. Ensure letdown remains in service while utilizing auxiliary spray (560°F Δ T limit). NOTE: During cooldown, actual PZR level can be determined by using the attached curve and indicated level at the dedicated shutdown panel (LI-406).
7. When RCS temperature has been reduced below 543°F (as indicated at the dedicated shutdown panel) open the following instrument AC breakers to inhibit HI steam line flow and Low Pzr pressure SI.	7. Automatic and manual SI is not available with these breakers open. If plant conditions warrant initiation of auto SI, close both instrument AC breakers.

ACTIONS	COMMENTS
a. PY 1115 Train A Output Cab.	
b. PY 1418 Train B Output Cab.	
8. Shutdown two RCP's to decrease the heat input to the system. Leave RCP #2 running, as it is connected to the PZR surge line.	8. Local indication of RCP vibration and RCP seal injection flows should be observed periodically during cooldown.
9. Do not allow the temperature difference between loops to exceed 25°F. The temperature of RCS loop 1 is directly indicated at the dedicated shutdown panel. Temperature for loops 2, 3 and 4 can be determined as a function of steam gen. pressure using the saturation curve on the attached graphs.	
10. When RCS pressure decreases below 1915# close the accumulator outlet valves (one at a time) by performing the following for each breaker:	10. MOV 8808A 52-1F-46 MOV 8808B 52-1G-07 MOV 8808C 52-1H-14 MOV 8808D 52-1G-05
a. Lay down close contactor seal in wire on terminal #4 (inside bkr cubicle).	
b. Close breaker.	
c. Depress close contactor.	
d. When close contactor drops out, immediately re-open breaker. (If open contactor picks up before breaker is opened, close valve fully using local handwheel.	d. Closure of these valves can be verified locally.
e. Re-lift and tape seal in wire on terminal #4.	

ACTIONS	COMMENTS
11. When RCS pressure drops below 1500# or RCP #1 seal leak-off flow decreases below 1 GPM (can be observed locally) open #1 seal bypass valve 8142 by placing pneumatic jumper around SV 205 at PM 90. (El. 91' East side of containment.)	
12. Continue to cool RCS until the temperature is less than 350°F and pressure is below 390#. Maintain pressure by reducing spray flow and energizing PZR heater.	12. Open additional orifices as necessary to maintain normal letdown flow.
13. Place RHR system in service on recirc. to collect a RHR system water sample for boron determination by manually closing MOV 8809A&B (RHR to C.L. 1,2,3 & 4) and starting RHR pumps from their switchgear cubicles. Observe that recirc. flow has been established (Observe FI's located locally outside the RHR pump rooms).	13. Cut in control transfer relays cutout switches on the RHR pumps prior to starting locally. Observe pump amps locally. Do not allow the RHR to run longer than 30 min. on recirc.
14. Collect RHR sample and analyze for boron conc. Ensure boron conc. equal to or greater than the RCS boron conc.	
15. Shutdown both RHR pumps.	
16. Close 8980 using handwheel.	16. RWST to RHR suction.
17. Close the breakers for 8701 (52-1G-25) & 8702 (52-1H-19).	17. RHR suction from RCS H.L.4
18. Open 8702 and 8701 by momentarily pushing the open contactor for the valve. The valve should be full open in about 100 seconds. When the valve is full open as indicated by the open contactor dropping out OPEN the valve breaker to prevent the valve from reclosing automatically.	18. Observe local pump suction pressure indication (which should increase to RCS pressure).

19. Open the CCW outlet FCV 364 and FCV 365 (RHR HX's) by shutting off the air supply to the valves. (Valves fail open on loss of air.)

19. Observe CCW flow thru HX's local FI's (100' El Aux. Bldg.)

20. Start #1 RHR pump on recirc.

20. Monitor recirc. flow locally.

21. Slowly open 8809A until an increase in flow is observed This is to allow the RHR system to heat up.

21. Observe local temperature indicator on RCS outlet of RHR HX. (located in RHR HX room)

22. Start #2 RHR pump as per steps 20 and 21 above. (except using 8809B in step 21).

23. Slowly open 8809A & B to commence further cooldown of RCS.

24. Install signal simulator units to PM 135 and FM 133 (located in PM 87 on 85' el. of Aux. Bldg.)

24. This allows for operation of HCV 133 and PCV 135. Contact I&C personnel for installation.

25. Install pressure gauge upstream of PCV 135 to monitor letdown press.

26. Open RHR Hx crosstie valves 8734A & B and establish RHR to letdown by fully opening HCV 133.

27. Regulate letdown flow by controlling PCV 135.

27. Letdown orifice valves should remain open.

28. When condenser vacuum can no longer be maintained, proceed as follows:

28. Condenser vacuum can be observed at the main unit turbine pedestal on PI 256, or on local indicator in PM 177 on 104' El. (near PY 17).

a. Terminate condenser steam dump (if in use).

ACTIONS	COMMENTS
b. Break condenser vacuum.	
c. Shutdown the air ejectors and secure shaft seal system on the main feed pumps and main unit turbine.	
29. Open & rack out SI pump breakers (52-HH-15 and 52-HF-15) and manually shut valve 8835. (SI to C.L. 1,2,3 and 4.)	
30. Reduce letdown flow less than charging flow to begin increasing PZR level to approx. 90%.	
31. When RCS temp. is reduced below 200°F, fill the steam generators all the way and place in wet layup.	
32. Ensure shutdown margin remains greater than 1.0% Δ K/K.	
33. Fill the PZR all the way.	33. While filling and cooling down the PZR with auxiliary spray, per Appendix B, ensure the PZR cooldown rate does not exceed 200°F/HR.
34. Shutdown remaining RCP's when the RCS temp. is below 160°F.	
35. Continue to cooldown the PZR until the PZR temp. is below 150°F.	
36. Reduce the system pressure to approx. 50 psi by reducing charging flow and increasing RHR letdown flow.	
37. Shutdown the charging pump.	

38. Stop any spray flow in progress and close the RHR letdown valve HCV 133.

39. Leave the RHR system in service recirculating from hot leg 4 to the cold legs.

40. After approximately 72 hours of cooling, one RHR train and related auxiliary systems may be shut down. The remaining RHR train must remain in service.

APPENDIX A

LETDOWN ISOLATION

Should letdown isolation occur due to low PZR level, proceed as follows:

1. Place control switch(s) for LTDN orifice valve(s) in CLOSED position.

2. Re-establish PZR level via charging flow control.

3. At nuclear auxiliary relay rack "B", depress and hold in energized position relays 33aoX / LCV 459 and 33aoX / LCV 460 for approx. 10 seconds.

   NOTE: Relays 33aoX / LCV 459 and 33aoX / LCV 460 are located in the top right hand corner of RNARB (cable spreading rm. 128' el.) and are appropriately labeled..

4. Open letdown orifice isolation valve(s) as required from the hot shutdown panel.

5. Check letdown flow re-established.

APPENDIX B

PRESSURIZER SPRAY ACTUATION

When pressurizer spray actuation is required in the performance of this procedure use method A or B below:

A. With RCP 1-1 or 1-2 operating.

1. This method uses a normal pressurizer spray valve and requires a RCP associated with its respective spray valve to be in operation.

a. To use PCV 455A (Loop 1):

(1) Contact the I&C Dept. and have them disconnect the output of PC 455G in Hagan Rack 19 at TB H leads #9 and #10, and install a 4-20 ma current source (Transmation model 1040 or equivalent) to the disconnected leads. This will allow for modulation of PCV 455A.

(2) Establish phone communications between the Hot Shutdown Panel and Hagan rack area.

(3) Increasing current to the valve will cause increased opening of the valve.

(4) When the valve is closed and operation of the valve is not required, turn off transmation unit.

b. To use PCV 455B (Loop 2)

(1) Contact the I&C Dept. and have them disconnect the output of PC 455F in Hagan Rack 19 at TB H leads #5 and #6, and install a 4-20 ma current source (Transmation model 1040 or equiv.) to the disconnected leads. This will allow for modulation of PCV 455 B.

(2) Establish phone communications between the Hot Shutdown Panel and Hagan rack area.

(3) Increasing current to the valve will cause increased opening of the valve.

(4) When the valve is closed and operation of the valve is not required, turn off transmation unit.

APPENDIX B CONT'D

B. With no RCP's in operation

1. This method requires access into containment on the 91 ft. elev. Proceed as follows:

   a. Contact the I&C Dept. and have them install a pneumatic jumper around SV 66 in Mech. panel 90 (located 91 ft. elev. next to the aux. spray valve) for aux. spray valve 8145. (Valve is air to open, fail closed).

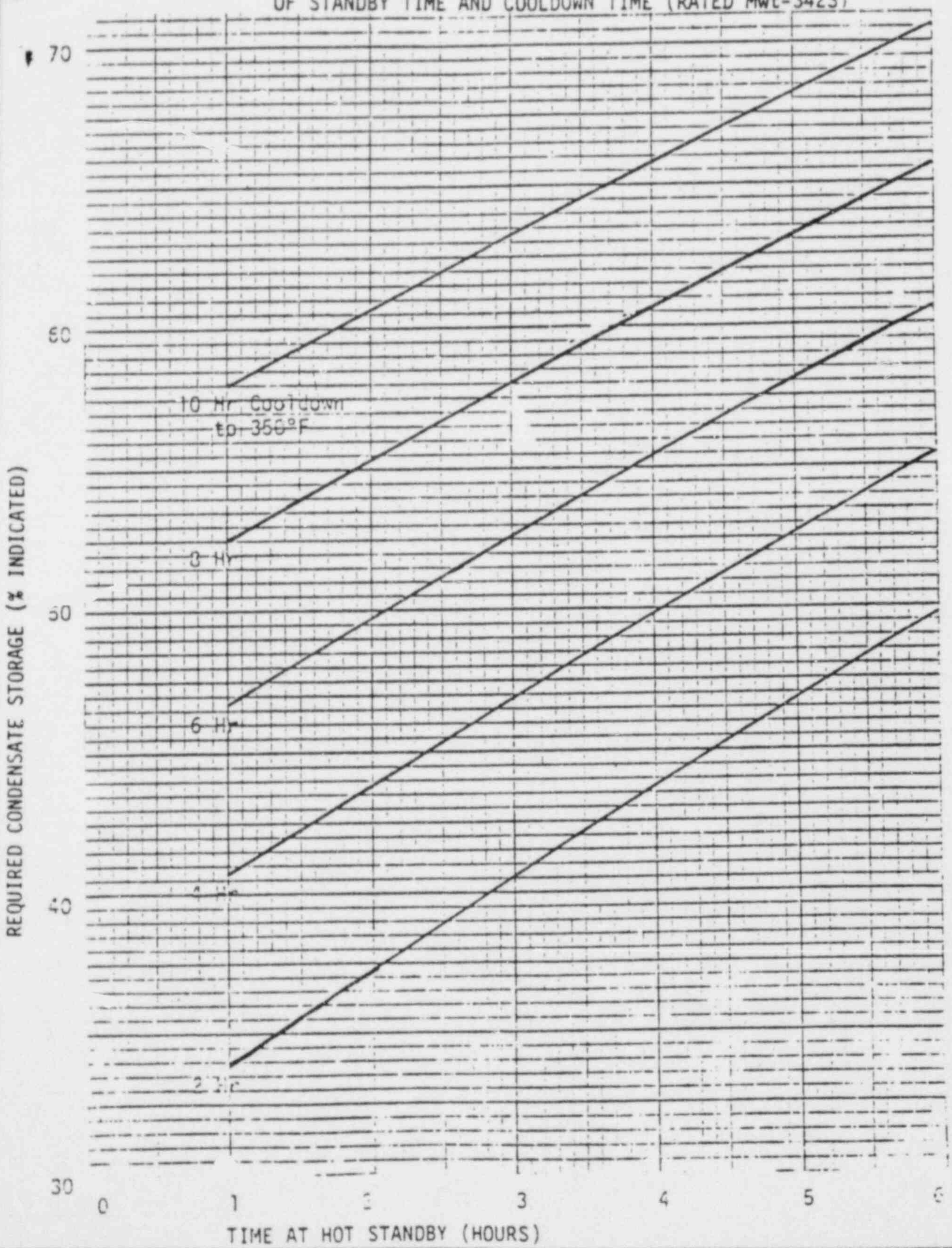
   b. Establish phone communications between Mech. panel 90 and the Hot Shutdown Panel.

   c. Cut in air to open valve 8145 and cut out air to close the valve as required.

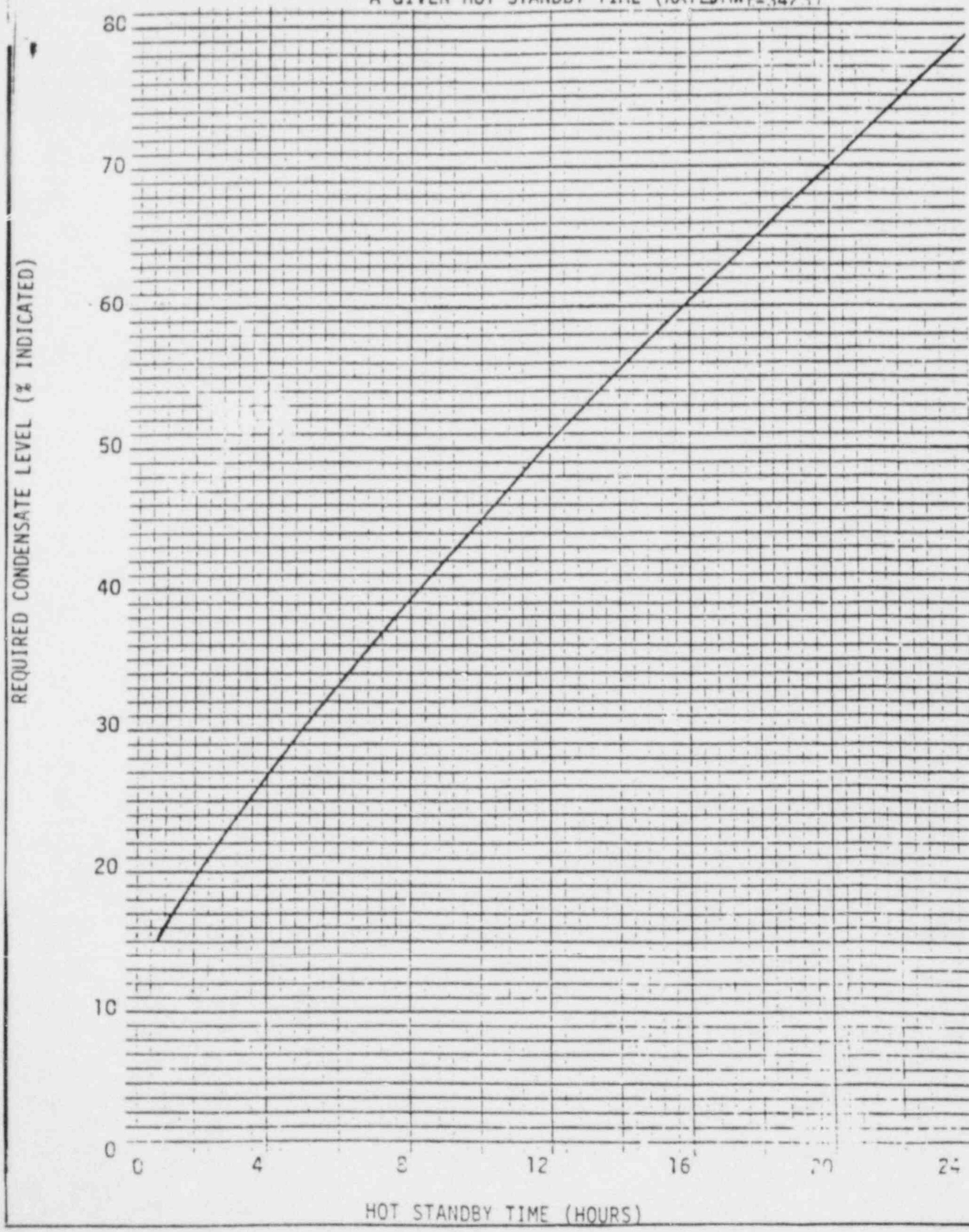
   d. Observe valve locally to ensure valve is responding correctly.

TITLE: CONTROL ROOM INACCESSIBILITY

REQUIRED CONDENSATE STORAGE CAPACITY AS A FUNCTION OF STANDBY TIME AND COOLDOWN TIME (RATED Mwt=3423)

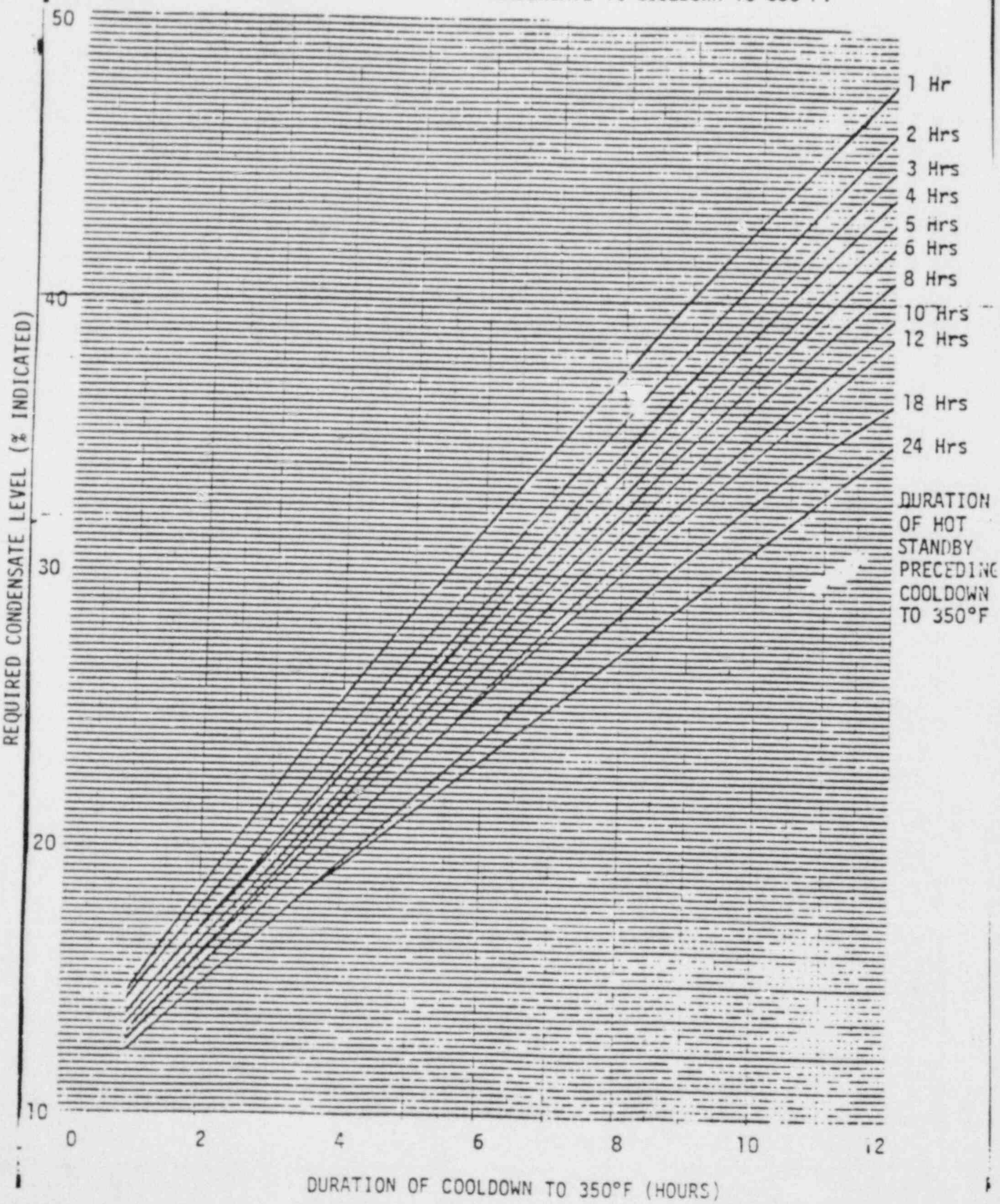


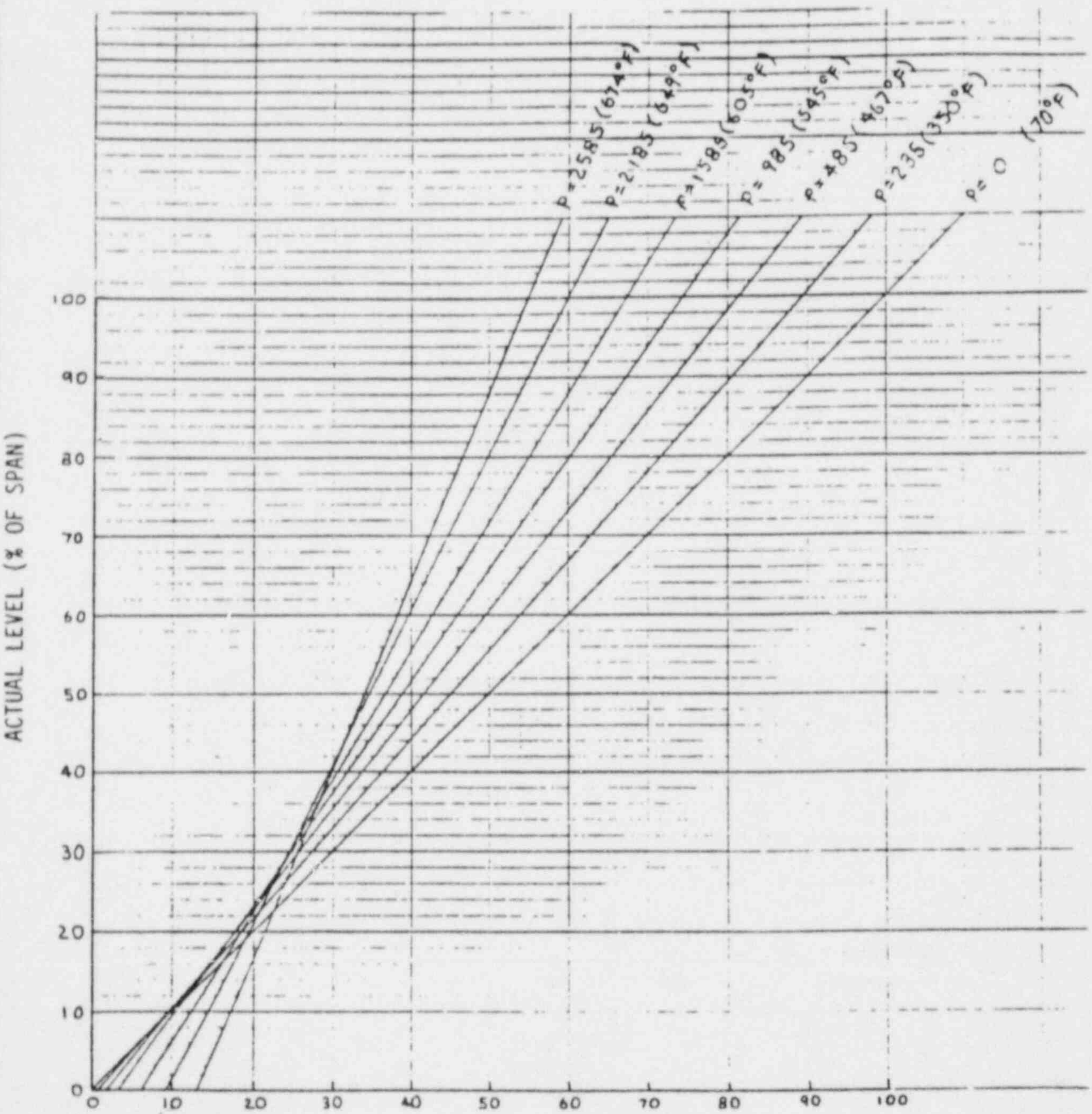
REQUIRED CONDENSATE LEVEL TO MAINTAIN
A GIVEN HOT STANDBY TIME (RATED MW_t=3423)



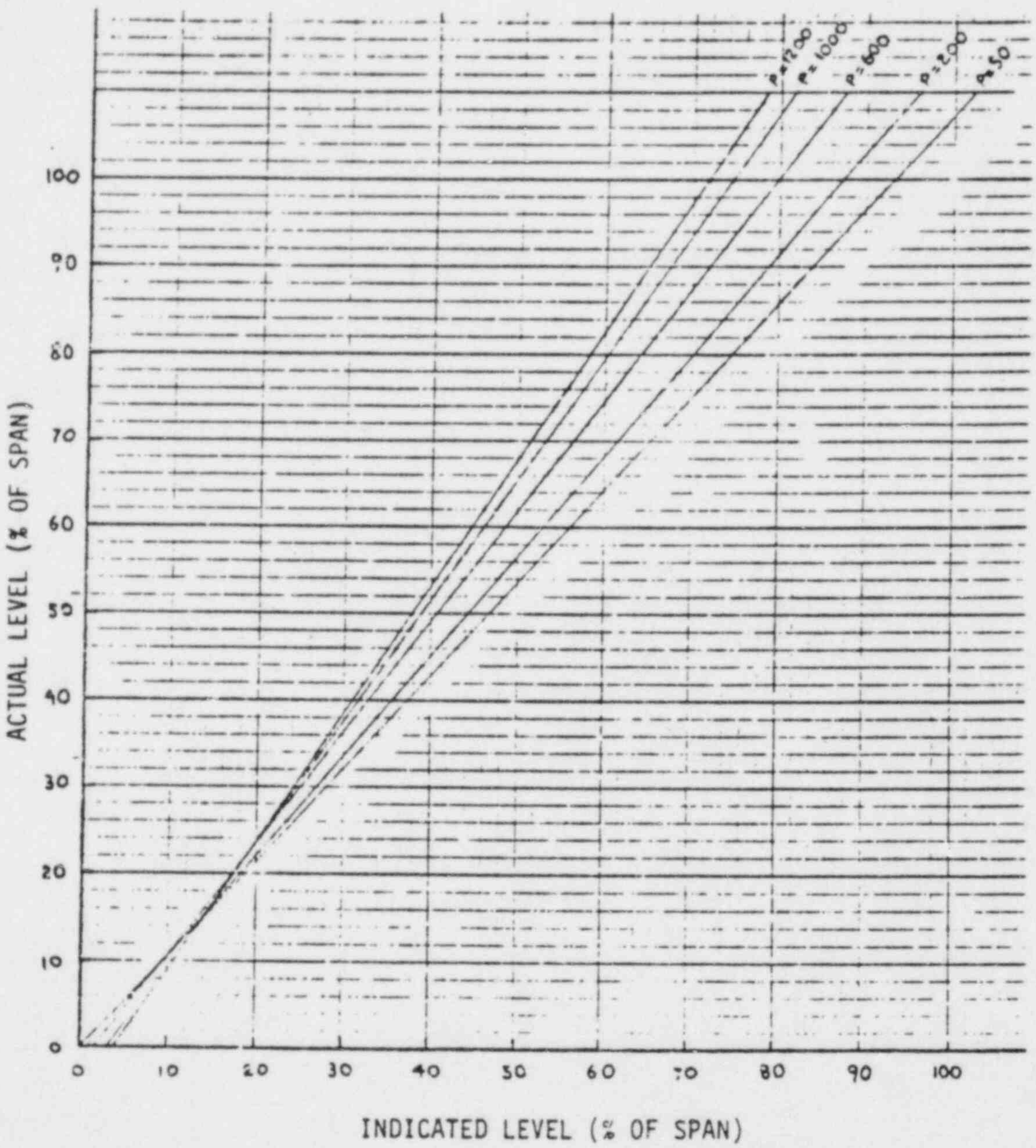
DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2	NUMBER EP OP-8 REVISION 5 DATE 2/11/82 PAGE 19 OF 23
TITLE: CONTROL ROOM INACCESSIBILITY	

REQUIRED LEVEL OF CONDENSATE TO COOLDOWN TO 350°F.





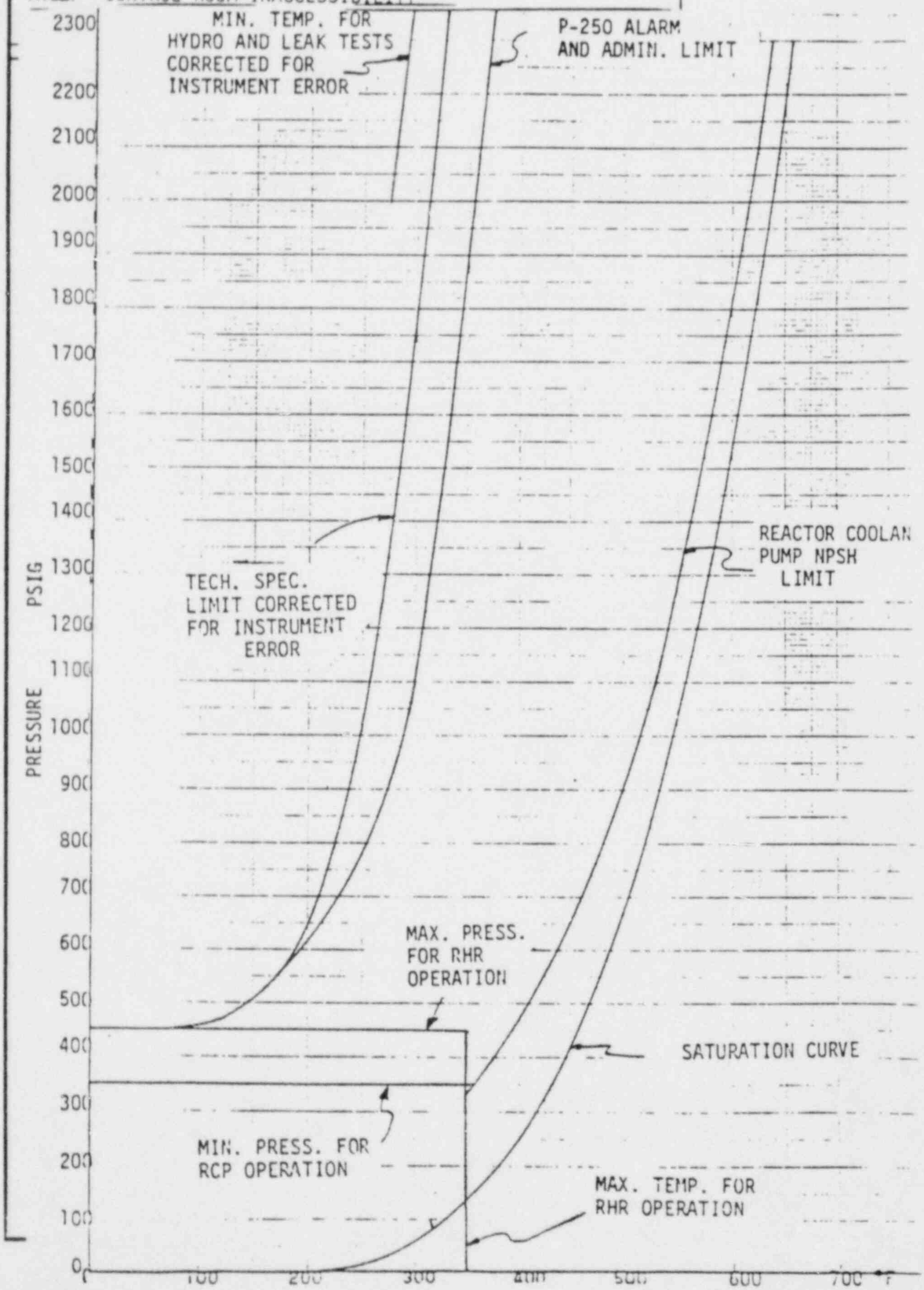
INDICATED LEVEL (% OF SPAN)
LI-406
PRESSURIZER LEVEL CORRECTION CURVES FOR PRESSURIZER PRESSURES (PSI
(CONTAINMENT TEMPERATURE=100°F EXCEPT AT : 0 PSIG; 70°F)



STEAM GENERATOR LEVEL (WIDE RANGE) CORRECTION CURVES
FOR STEAM GENERATOR PRESSURE (PSIG)
(CONTAINMENT TEMPERATURE-100°F)

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2	NUMBER	EP OP-8
	REVISION	5
	DATE	2/11/82
	PAGE	22 OF 23

TITLE: CONTROL ROOM INACCESSIBILITY



APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Designate this event an Alert. Notify plant staff and response organizations required for this classification by implementing Emergency Procedure G-2 "Establishment of the On-Site Emergency Organization" and G-3 Emergency Procedure "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

   b. If, after evacuation of the Control Room, control of shutdown systems cannot be established within 15 minutes, redesignate this event as a Site Area Emergency. Notify plant staff and response organizations required by EP G-2 and EP G-3 in accordance with EP G-1.

TITLE: EMERGENCY OPERATING PROCEDURE
LOSS OF A REACTOR COOLANT PUMP

APPROVED: R. C. Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure provides instructions to follow in the event of a loss of a reactor coolant pump during power operation. Two separate conditions are covered relating to whether or not the loss of the pump results in a reactor trip. Operation above the P-8 setpoint produces a reactor trip on the loss of one coolant pump while operation above the P-7 setpoint results in a reactor trip only if two coolant pumps are lost. The procedure is the same regardless of whether if takes one or two pumps to trip the reactor.

SYMPTOMS

1. Low coolant flow and pump trip alarm (PK04-11 or 12).
2. Coolant pump breaker trip and alarm (PK04-11 or 12).
3. Coolant loop ΔT deviation alarm (PK04-01).
4. Level in one steam generator rapidly decreasing and levels in the other steam generators rapidly increasing.
5. Steam flow in one steam line rapidly decreasing and steam flow in the other three lines increasing.

AUTOMATIC ACTIONS

1. Possible reactor trip.

OBJECTIVES

1. Stabilize plant conditions.

IMMEDIATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. With a reactor trip:	
a. Check that the reactor has tripped and manually trip if it has not.	a. Verify by insuring all rods are in, neutron flux level is decreasing and a negative SUR is observed on the IR/SR rate meters.

ACTIONS	COMMENTS
b. Check that the turbine has tripped and manually trip if necessary.	b. Verify by insuring stop valves are closed.
c. Verify that a heat sink exists.	c. Verify by insuring that steam dump is occurring, T_{AVG} is approaching 547°F and feedwater is being supplied to the steam generators.
d. Follow the reactor trip procedure OP-5.	

2. Without a reactor trip:

   a. Check steam generator levels and place the level control for the steam generator with the failed pump on MANUAL and stabilize the level.

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. With a reactor trip:	
a. Check pressurizer pressure and level being maintained at no load conditions.	
b. Check remaining pumps for satisfactory operation.	b. Check indicators on VB-2: pump amps, seal ΔP, seal temperatures and seal inlet flow. Check annunciator for RCP vibration (PK05-5) and the other RCP main annunciators (PK05-1, 2, 3, or 4).
c. Refer to Subsequent Actions of Emergency Operating Procedure No. OP-5.	

2. Without a reactor trip:

   a. With steam generator level stabilized, attempt to return affected steam generator to automatic level control.

ACTIONS

COMMENTS

b. Within one hour place the bistables for the affected loop associated with the following ESF features in the trip position:

1) Low-Low T_{AVG} signal used in the safety injection signal.

2) Low steam line pressure signal used in the safety injection signal.

3) High steam flow signal used for safety injection.

4) Steam line high ΔP signal used for safety injection. (Trip all bistables which indicate low active loop pressure relative to idle loop pressure.)

c. If the cause for the pump trip has been corrected and the pump is to be placed back in operation, refer to Operating Procedure No. A-6, "Start On An Idle Pump".

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this a significant event and as a minimum within one hour notify the NRC Bethesda Operations Center using the red phone ·r the Control Room. Gather sufficient information from all sources pr or to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CRF Part 50.72, (Notification of Significant Events).

   c. When the cause of the reactor coolant pump trip has been determined, this and the subsequent plant conditions should be reviewed against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plant Activation."

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-10
REVISION 2
DATE 2/11/82
PAGE 1 OF 3

TITLE: EMERGENCY OPERATING PROCEDURE
LOSS OF AUXILIARY SALT WATER

APPROVED: R C Thornberry 2/9/82
PLANT MANAGER DATE

SCOPE

This procedure covers the steps to be taken following a loss of auxiliary salt water. This system operates to remove waste heat from the component cooling water system during both normal and accident conditions.

SYMPTOMS

1. Auxiliary salt water header low pressure - alarm.
2. Possible auxiliary salt water pump overcurrent trip - alarm.
3. Component cooling water heat exchanger water box differential pressure - alarm.
4. Component cooling water heat exchanger CCW outlet temperature high - alarm.
5. Possible auxiliary salt water pump screen differential high - alarm.

AUTOMATIC ACTIONS

1. Automatic start of standby auxiliary salt water pump.
2. Possible automatic start of screen wash system.

OBJECTIVES

1. Maintain adequate cooling capacity to CCW heat exchangers during normal and upset conditions.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Verify that the standby auxiliary salt water pump started automatically and that flow has been re-established. Start the pump manually if it did not start automatically.	1. Observe motor amp meter.
2. If the standby pump will not start or flow cannot be re-established, open the cross tie valve FCV-601 between Units 1 and 2. Start alternate standby pump.	

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. If the ASW pump (or pumps) are running and flow does not recover or CCW temperature remains high, proceed to change over CCW heat exchangers.	1. See Operating Procedures E-5 and F-12.
2. If the loss of flow resulted from a blocked screen causing loss of suction, proceed to restore by opening the demusseling valves FCV-432 and FCV-433.	
3. Take appropriate steps to return the system to normal as soon as practical.	
4. Notify appropriate plant supervision.	
5. Refer to Technical Specifications to determine the time available for repairs if equipment has failed.	
6. Continue to operate at power if CCW temperature is normal.	
7. If CCW temperature remains high, start a controlled load reduction to reduce the heat load on the system. Shutdown auxiliary equipment ot the extent possible.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Designate this event as a Notification of Unusual Event. Notify plant staff and response organizations required for this classification by implementing Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

   b. If flow cannot be established in accordance with step 2, designate the event as follows:

      Plant in modes 5 or 6: Alert if the event results in a loss of Residual Heat Removal System functions.

      Plant in modes 1, 2, 3 or 4: Site Area Emergency

   Notify plant staff and response organizations required by EP G-2 and EP G-3 in accordance with EP G-1.

TITLE: EMERGENCY OPERATING PROCEDURE
FAILURE OF A CONTROL BANK TO MOVE IN AUTOMATIC

APPROVED: R C Thornberry 2/13/82
PLANT MANAGER DATE

SCOPE

This procedure provides uniform instructions for plant operation, when the control rod system fails to respond to a TAVG - TREF mismatch while in automatic mode.

SYMPTOMS

This condition may be indicated by one or more of the following:

1. Rods fail to withdraw on a turbine load increase or while borating the primary system.

   a. TAVG - TREF deviation alarm (PK04-03).

   b. Rod insertion low or low-low limit alarm (PK03-13 or 14).

   c. PZR lvl hi/low control alarm and letdown isolation (PK05-22).

   d. Decreasing TAVG indication.

   e. PZR pressure low (PK05-17).

   f. PZR level hi/low alarm (PK05-21).

   g. Bank D rod stop C-11 (PK03-15).

2. Rods fail to insert on a turbine load decrease or while diluting the primary system.

   a. TAVG - TERF deviation alarm (PK04-03).

   b. Increasing TAVG and a possible a...t. TAVG hi alarm (PK04-10).

   c. High pressurizer pressure alarm if the condition persists (PK05-16).

   d. PZR lvl hi/low control alarm and B/U heaters on if the condition persists (PK05-22).

3. Rod urgent failure alarm (PK03-17).

4. Possible OTΔT or OPΔT rod withdrawal stop and turbine runback status windows (PK08-09 or 10) and alarms (PK04-04 or 05).

AUTOMATIC ACTIONS

1. During an increase in turbine load or boron concentration the following may occur.
   a. Pressurizer heaters will energize.
   b. CVCS letdown isolation.
2. During a decrease in turbine load or boron concentration the following may occur.
   a. Actuation of pressurizer sprays.
   b. Possible actuation of the pressurizer power relief valves.
   c. Possible OTΔT or OPΔT rod stop and turbine runback.
   d. Possible OTΔ or OPΔT reactor trip.

OBJECTIVES

1. Adjust T_{AVG} or T_{REF} so there is no mismatch.
2. If the control rod bank is inoperable, initiate an orderly shutdown.

IMMEDIATE OPERATOR ACTIONS

COMMENTS

1. Stop the operation in progress:
   a. Turbine load change.
   b. Boration.
   c. Dilution.
2. Match T_{AVG} - T_{REF} by using any of the available methods below.

COMMENTS

2. An urgent alarm in the logic cabinet inhibits all rod movement in any mode.

ACTION	COMMENTS
a. Readjust rods in manual. b. Readjust turbine load. c. Borate or dilute with the control rods in manual.	An urgent alarm in a power cabinet inhibits rod movement in auto or manual. Rod movement on individual selection is inhibited in affected cabinet only. Attempt to maintain ΔI in target band while performing the above.

SUBSEQUENT OPERATOR ACTION

ACTION	COMMENTS
1. If the control rods are functional in manual, operation may continue indefinitely.	
2. If the control rods cannot be moved in manual, use individual bank select and initiate system repairs.	2. The bank overlap control unit will not see any movement taking place in the individual bank control. Therefore bank D must be maintained above the overlap setpoint (100 steps withdrawn) during power operation. If this mode of control is used, reset the Bank Overlap Unit to the current bank position prior to transferring rod control to manual or auto mode.
3. If rods will not move in individual bank select or the control rod system is declared inoperable, initiate a controlled shutdown by borating the primary system to a minimum power level and then trip the reactor and follow OP-5 (Reactor Trip Without Safety Injection).	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman proceed as follows:

   a. Notify the Plant Superintendent, the Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this event a Significant Event. As a minimum, within one hour notify the NRC Bethesda Operations Center using the red phone in the control room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to the NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. The cause of the failure of the control rod system to respond in automatic, and the subsequent plant conditions, should be reviewed against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
CONTINUOUS WITHDRAWAL OF A CONTROL ROD BANK

APPROVED: [signature] 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure provides instructions for plant operation with the occurrence of the continuous withdrawal of a control rod bank while at power or approaching critical.

SYMPTOMS

This condition may be indicated by one or more of the following:

1. Unwarranted rod motion.
2. TAVG-TREF deviation alarm (PK04-03).
3. Increase in reactor power without a corresponding increase in turbine power.
4. Auct. TAVG hi alarm (PK04-10).
5. High pressurizer pressure alarm (PK05-16).
6. PZR lvl hi/low control alarm and B/U heaters on (PK05-22).

AUTOMATIC ACTIONS

1. Possible high nuclear flux reactor trip. (Source, Intermediate or Power Range.)
2. Possible overpower or overtemperature ΔT reactor trip.
3. If no reactor trip occurs, one or more of the following may occur:
   a. Power range high rod stop (C2).
   b. Intermediate range high rod stop (C1).
   c. Overpower ΔT rod stop and turbine runback (C4).
   d. Overtemperature ΔT rod stop and turbine runback (C3).
   e. Pressurizer spray or power relief actuation.

f. Bank D rod stop at 220 steps.

OBJECTIVES

1. Stop inadvertent withdrawal of the control rod bank and readjust T_{AVG} with T_{REF}.
2. To perform a controlled shutdown if the control rod bank is inoperable.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Transfer the rod control system to manual.	
2. If control rod motion does not stop, trip the reactor and refer to OP-5.	
3. If control rod motion stops, match TAVG with TREF by inserting control rods or increasing turbine load.	

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. If the control rod bank responds normally in manual continue power operation.	
2. If the control rods cannot be moved in manual, use individual bank select and initiate system repairs.	2. The bank overlap control unit will not see any rod movement in the individual bank control mode. Therefore, bank D must be maintained above the overlap setpoint (100 steps withdrawn) during power operation. If this mode of control is used, reset the bank overlap unit to the current bank position prior to transferring rod control to manual or auto mode.

TITLE: CONTINUOUS WITHDRAWAL OF A CONTROL ROD BANK

ACTION	COMMENTS
3. If rods will not move in individual bank select or the control bank is declared inoperable, initiate a controlled shutdown by borating the primary system to a minimum power level. Trip the reactor and follow OP-5.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman proceed as follows:

   a. Notify the Plant Superintendent, the Supervisor of Operations and the Plant Manager or their designated alternates.

   b. Designate this event a Significant Event. As a minimum, within one hour notify the NRC Operations Center using the red phone in the control room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to the NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. The cause of the control bank malfunction, and the subsequent plant conditions, should be reviewed against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-12C

REVISION 2

DATE 2/11/82

TITLE: EMERGENCY OPERATING PROCEDURE
CONTINUOUS INSERTION OF A CONTROL ROD BANK

APPROVED: R C Thornberry 2/15/82
PLANT MANAGER DATE

SCOPE

This procedure provides uniform instructions for plant operation with a continuous insertion of a control rod bank.

SYMPTOMS

This condition may be indicated by one or more of the following:

1. Unwarranted rod motion.
2. TAVG - TREF deviation alarm (PK04-03).
3. Rod insertion limit alarm (PK03-13 or 14).
4. Decreasing TAVG indication.
5. Possible pressurizer pressure low alarm (PK05-17).
6. Possible pressurizer lvl hi/low alarm (PK05-12).
7. Pressurizer lvl hi/low control and letdown isolation (PK03-22).
8. Possible P250 reactor alarm due to ΔI limits exceeded (PK03-25).
9. A reactor power decrease without a corresponding decrease in turbine power.

AUTOMATIC ACTIONS

1. Actuation of pressurizer heaters.
2. Possible CVCS letdown isolation and pressurizer heaters off.
3. Possible low pressurizer pressure reactor trip.

OBJECTIVES

1. Stop inadvertent insertion of control rod bank and readjust TAVG with TREF.
2. To perform a controlled shutdown if the control rod bank is inoperable.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Transfer the rod control system to manual.	
2. If control rod motion does not stop, trip the reactor and refer to OP-5.	
3. If the control rod motion stops, match T_{AVG} with T_{REF} by withdrawing control rods or decreasing turbine load.	

SUBSEQUENT OPERATOR ACTION

ACTION	COMMENTS
1. If the control rod banks respond normally in manual, continue power operation.	
2. If the control rods cannot be moved in manual, use individual bank; select and initiate system repairs.	2. The bank overlap control unit will not see any rod movement in the individual bank control mode, therefore, bank D should be maintained above the overlap setpoint (100 steps withdrawn) during power operation. If this mode is used, reset the Bank Overlap Unit to the current bank position prior to transferring rod control back to manual or auto mode.
3. If rods will not move in individual bank select or the control rod bank is declared inoperable, initiate a controlled shutdown by borating the primary system to minimum power level. Trip the reactor and follow OP-5.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman proceed as follows:

   a. Notify the Plant Superintendent, the Supervisor of Operations and the Plant Manager or their designated alternates.

   b. Designate this event a Significant Event. As a minimum, within one hour notify the NRC Operations Center using the red phone in the control room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events). Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting to the NRC)" for a suggested format for reporting.

   c. The cause of the control bank malfunction, and the subsequent plant conditions, should be reviewed against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-12D

REVISION 3

DATE 2/11/82

TITLE: EMERGENCY OPERATING PROCEDURE
CONTROL ROD POSITION INDICATION SYSTEM MALFUNCTION

APPROVED: R C Thornberry 2/15/82
PLANT MANAGER DATE

SCOPE

This procedure provides uniform instructions for plant operation when a malfunction of the rod position indication (RPI) system occurs.

SYMPTOMS

This condition may be indicated by any of the following:

1. One or more rod position indicators in disagreement with the group step counter or other RPI indicators by more than 12 steps.

2. Disagreement between the group step counters of the same bank by more than one step.

3. Any one or more of the following alarms:

   a. Rod position deviation (p250 RX alarm PK03-25).

   b. Central control failure (RPI panel).

   c. Data channel A failure (RPI panel).

   d. Data channel B failure (RPI panel).

   e. Rod position indicator system alarm (PK03-21)

AUTOMATIC ACTIONS

None

OBJECTIVES

1. Determine if an RPI failure has occurred or if a control rod misalignment exists.

2. Satisfy Technical Specifications action statement for particular condition.

TITLE: CONTROL ROD POSITION INDICATION SYSTEM MALFUNCTION

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Stop any load change in prog ess and allow conditions to stabilize.	
2. If the RPI panel is indicating a control rod misalignment and neutron flux, TAVG or ΔT deviations exist refer to OP-12E.	
3. If failures in the central control and/or the data channels exist on the DRPI panel, proceed to subsequent operator actions.	3. The "GW" alarm light will be lit for the effected rod(s). Also the rod bottom light will be lit if both data A & B channels have failed.

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. Place rod control system in manual.	
2. With a maximum of one rod position indicator per Bank inoperable [i.e., both data A and B channels failed on the rod(s)]. Refer to Technical Specification Action statement 3.1.3.2.(a).	2. Rod position can be confirmed as follows. a. Take an axial flux scan of each rod having a failed position indicator by using the moveable incore detectors. b. Perform a similar scan with the moveable incore detectors on a control rod with an operating rod position indicator which has a similar rod to detector geometry. Compare the traces to confirm the position of the affected rod.
3. With a maximum of one step counter per Bank inoperable, refer to Tech Spec Action Statement 3.1.3.2(b).	
4. Loss of data A and B channels of indication for any rod not fully inserted in Modes 3,4 and 5 requires immediate opening of RX trip breakers.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman proceed as follows:

   a. Notify the Plant Superintendent, the Supervisor of Operations and the Plant Manager or their designated alternates.

   b. Designate this event a Significant Event. As a minimum, within one hour notify the NRC Operations Center using the red phone in the control room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting to the NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. The cause of the control bank malfunction, and the subsequent plant conditions, should be reviewed against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

PG&E **Pacific Gas and Electric Company**

NUMBER EP OP-13
REVISION 2
DATE 2/10/82
PAGE 1 OF 9

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

TITLE: EMERGENCY OPERATING PROCEDURE
MALFUNCTION OF REACTOR PRESSURE CONTROL SYSTEM

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

The purpose of this procedure is to provide instructions to be followed in the event of a malfunction of the reactor pressure control system. For simplicity, this procedure is subdivided into four parts as follows:

PART A: Pressurizer Pressure Channel fails high.

PART B: Stuck open spray valve.

PART C: Pressurizer heater malfunction.

PART D: Stuck open or leaking power operated relief valve.

PART A: PRESSURIZER PRESSURE CHANNEL FAILS HIGH

SYMPTOMS

1. Failed channel will indicate high pressure. Give the high pressurizer pressure alarm (PK05-16) and trip the associated reactor trip bystable (PK05-6).

2. Other pressure channels show actual pressure is decreasing and gives low pressure alarm (PK05-17).

3. PRT pressure, level and temperature indicators on VB-2 show abnormally high indication.

4. "Pressurizer PORV Temperature High" annunciator alarms (PK05-23).

AUTOMATIC ACTIONS

1. If the controlling channel fails high.

   a. Both spray valves open.

   b. All pressurizer heaters de-energized.

   c. PCV-455C opens (closes again when pressure decreases to 2185 psig).

2. If the selected protection channel fails high, PCV-474 and PCV-456 will open (but close when actual pressure decreases to 2185 psig).

3. Possible reactor trip on low pressure.

4. Possible safety injection signal on low pressure.

OBJECTIVES

1. To regain pressure control by selecting the alternate control channel to terminate event without a reactor trip.

2. To restore p ̣essurizer pressure and level to reference values.

3. To trip all the bistables associated with the fa.led channel within one hour.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Select alternate channel for pressure control on selector switch P/455A.	1. Insure that the position selected does not utilize the defective channel.
2. Verify spray valves closed, PORV's closed and heaters energized.	
3. If step 2 does not occur, place the master pressure controller (HC-455K) in manual and return pressure to normal.	
4. Stop any unit load changes in progress.	
5. If a reactor trip occurs, refer to Emergency Operating Procedure No. OP-5	
6. If a safety in ction occurs, refer to Emergency Operating Procedure No. OP-0.	

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. Select the pressurizer pressure recorder to record a valid pressure channel (P/455B).	

ACTION	COMMENTS
2. If a reactor trip does not occur:	
a. Ensure pressurizer level and pressure return to their normal value.	
b. Within one hour trip, all the by-stables associated with the failed pressurizer pressure channel.	b. See Table III A-1 in Volume 9 of DCPP Plant Manual.
3. Refer to Operating Procedure No. A-4B to return the PRT to normal.	

PART B: STUCK OPEN SPRAY VALVE

SYMPTOMS

1. All pressurizer pressure channels show decreasing pressure.
2. Low pressurizer pressure alarm (PK05-17).
3. Pressurizer heaters and spray on at the same time.
4. Pressurizer surge line temperature is lower than normal due to upsurge and will give an alarm.

AUTOMATIC ACTIONS

1. Backup heater energize.
2. Possible reactor trip on low pressure.
3. Possible safety injection signal.

OBJECTIVES

1. To terminate spray valve action.
2. To restore pressure control.

IMMEDIATE OPERATOR ACTION

ACTION	COMMENTS
1. Place affected spray valve controller to manual and close the spray valve.	
2. If spray valve does not completely close, ensure all heater groups are energized.	
3. Quickly drop load, if necessary, to reduce the power level to about 20% Then trip the reactor coolant pump of the loop associated with the defective spray valve.	3. If the RCP is tripped at a power level much greater than 20%, the reactor will probably trip on low steam generator level. If the RCP is tripped above 35% power, the reactor will trip. NOTE: Loop No. 1 feeds PCV-455A and Loop No. 2 feeds PCV-455B.
4. If step 3, above, results in a reactor trip, then: a. Trip the reactor coolant pump of the loop associated with the affected spray valve. b. Proceed to Emergency Operating Procedure No. OP-5.	
5. If a safety injection occurs, proceed to Emergency Operating Procedure No. OP-0.	

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. If system pressure was maintained without a reactor trip. a. Return system to normal pressure by manual heater operation.	a. Use the minimum number of heater banks to maintain pressure.

ACTION	COMMENTS
b. If normal spray is not useable or inaffective when system pressure is restored, use auxiliary spray or use a pressurizer relief. Follow step No. 1C or 1D below.	
c. If auxiliary spray is needed.	
1) Ensure letdown is not isolated.	1) If letdown is isolated or becomes isolated, auxiliary spray cannot be used due to the ΔT limit (Tech Spec 3.4.9.2).
2) Close the normal charging path to the reactor coolant loop.	
3) Monitor and maintain adequate RCP seal injection.	
4) Open auxiliary spray valve to control pressure.	
d. If a pressurizer relief is needed for pressure control,	
1) Close one pressurizer relief isolation valve.	
2) When the isolation valve is closed, open its' associated PORV.	
3) The motor operated isolation valve can now be jogged for pressure control.	
e. If a spray valve is stuck open, evaluate the conditions necessary for containment entry to repair or isolate the defective valve.	
f. If the defective valve was closed, repair if possible or evaluate plant conditions to determine the feasibility of continued power operation in such a mode.	

PART C: PRESSURIZER HEATER MALFUNCTION

SYMPTOMS

The following are symptoms for 3 possible failures:

1. If controlling pressurizer heater channel fails low.

   a. Low pressurizer pressure and alarm (PK05-17) on defective channel.

   b. High pressurizer pressure and alarm (PK05-16) on other channels.

   c. Pressurizer high temperature alarm (PK05-18).

   d. No automatic initiati n of sprays.

2. If heaters fail to de-energize.

   a. Pressurizer sprays and heater operating simultaneo ly.

3. Loss of pressurizer heaters.

   a. Low pressurizer pressure on all channels and alarm (PK05-17).

   b. No automatic initiation of pressurizer heaters.

AUTOMATIC ACTIONS

1. If controlling channel fails low, backup heaters will energize and soon pressurizer power operated relief valves will open when actual pressurizer pressure reaches 2335 psig.

2. If the heaters fail to deenergize automatically, pressurizer sprays will actuate to maintain pressure.

OBJECTIVES

1. To maintain pressure with manual control of heaters if it is determined that the pressurizer heaters are malfunctioning.

2. To reestablish automatic pressure control by selecting the alternate pressurizer control channel.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Take manual control of heaters and restore pressure to normal setpoint value.	
2. If the controlling pressurizer pressure channel failed low, select the alternate control channel and reestablish normal auto pressure control	
3. Stop any unit load change in progress	

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. With a failed pressurizer pressure channel, within one hour trip all the bistables associated with the failed pressurizer pressure channel.	

PART D: STUCK OPEN OR LEAKING POWER OPERATED RELIEF VALVE

SYMPTOMS

1. Low pressurizer pressure indication and alarm.
2. Pressurizer relief tank pressure, level and temperature indicators read high (VB-2).
3. Pressurizer relief valve discharge temperature indicator reads high (VB-2).
4. Pressurizer PORV high temperature alarm (PK05-23).

AUTOMATIC ACTIONS

1. Backup heaters energize.
2. Reactor trip on low pressurizer pressure.
3. Safety injection on low pressurizer pressure.

IMMEDIATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. If the defective relief valve can be identified by position lights, close its motor operated isolation valve.	
2. If the defective valve cannot be identified by position lights, close all three motor operated PORV isolation valves.	
3. If the reactor trips, refer to Emergency Operating Procedure No. OP-5.	
4. If a safety injection occurs, refer to Emergency Operating Procedure No. OP-0.	

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. If the defective relief valve is isolated without a reactor trip, restore pressurizer pressure and level to reference values.	
2. If the defective valve was not identified:	
a. Wait until the discharge header temperature decreases.	
b. One at a time, crack open a motor operated isolation valve and identify the defective PORV.	b. Identify by using the discharge header temperature, PRT conditions and RCS pressure fluxuations.
c. When identified, close the associated isolation valve at the faulted PORV.	
d. Open the isolation valves on the unaffected relief valves.	
3. With abnormal conditions in the PRT, refer to Operating Procedure No. A-4B.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this event a Significant Event. As a minimum, within one hour notify the NRC Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to the NRC)" for a suggested format for this report. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. If the pressure transient results in one of the following:

      1) Over temperature ΔT or Overpower ΔT protection channel activated.

      2) Low pressurizer pressure alarm and High T_{avg} alarm simultaneously.

      3) Failure of a relief or safety valve to close.

      Designate this event a Notification of Unusual Event. Notify Plant staff and response organizations required for this classification by implementing procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
NUCLEAR INSTRUMENTATION MALFUNCTIONS

APPROVED: [signature] 2/15/82
PLANT MANAGER DATE

SCOPE

This procedure covers the steps to be taken following a failure of any nuclear instrumentation channel. Each type of channel is considered separately in this procedure, as follows:

A. SOURCE RANGE CHANNELS

B. INTERMEDIATE RANGE CHANNELS

C. POWER RANGE CHANNELS

A. SOURCE RANGE CHANNELS N-31 AND N-32

Symptoms

1. Loss of indication.
2. High flux at shutdown alarm (PK03-07).
3. Loss of detector voltage alarm (PK03-06).
4. Audio count rate signal stops.
5. Possible source range high flux trip alarm.

AUTOMATIC ACTIONS

1. Possible reactor trip.
2. Possible containment evacuation alarm.

OBJECTIVES

1. Prevent undetectable reactivity or power changes.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. If the reactor tips follow the trip procedure.	
2. If the reactor is shutdown and the hi flux at shutdown is initiated:	

ACTION	COMMENTS
a. Notify personnel in containment.	
b. Determine whether or not alarm was spurious.	
3. If in mode 6 and a monitor fails or visual or audible indication is lost, immediately suspend all operations involving core alterations or positive reactivity changes.	
Subsequent Operator Actions	
1. If in mode 6 and a second channel fails, determine the boron concentration of the RCS at least once per 12 hours.	
2. If in mode 6 and one channel failed, select the operable channel for both audible indication and scaler operation.	
3. In modes 3, 4 or 5:	
a. If no channels are operable, verify adequate SDM exists within one hour and at least once per 12 hours thereafter.	a. Tech Specs 3.1.1.1 or 3.1.1.2
b. When the reactor trip breakers are closed, open the trip breakers if any source range channel becomes inoperable.	
4. During mode 2, and one channel fails:	
a. Below P-6, restore channel to operable status prior to increasing thermal power above P-6.	
b. Above P-6, continue startup on intermediate range channels.	
5. During mode 2 and both channels fail, follow the requirements of Tech. Spec. 3.0.3 and perform step 3.a. above.	

B. INTERMEDIATE RANGE CHANNELS N-35 AND N-36

Symptoms

1. Loss of channel indication.
2. Possible intermediate range high flux trip
3. Possible intermediate range high level rod stop.
4. Loss of compensating voltage alarm (PK03-03).
5. Loss of detector voltage alarm (PK03-03).

AUTOMATIC ACTIONS

1. Possible reactor trip.
2. Possible rod withdrawal block.
3. Possible loss or actuation of P-6.

OBJECTIVES

1. Prevent undetectable reactivity or power changes.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. If the reactor trips, follow the trip procedure.	

SUBSEQUENT OPERATOR ACTIONS

1. When one channel fails:

   a. If below P-6, restore channel prior to increasing thermal power above P-6.

   b. If reactor power is above P-6 but less than 5%, bypass the trip and maintain less than 5% power. Restore the inoperable channel prior to entry into mode 1.

   c. If reactor power is above 5%, operation may continue.

   d. If in modes 3, 4 or 5 with the reactor trip breakers closed, open the trip breakers.

2. If channel fails or a P-6 interlock channel becomes inoperable follow the additional requirement of Tech. Spec. 3.3.1, Action #8a.

3. If both intermediate range channels fail and power is <10%, follow the requirements of Tech. Spec. 3.0.3. If above 10%, Specification 3.0.3 is not applicable. Follow additional requirements of Tech. Spec. 3.3.1, Action #8a.

4. Loss of compensating voltage on one or both channels following a reactor trip will require manual reset of source range channels.

C. POWER RANGE CHANNELS N-41, N-42, N-43 & N-44

SYMPTOMS

1. Loss of indication.
2. Loss of detector voltage alarm (PK03-06).
3. Power range rod stop alarm (PK-3-09).
4. Channel deviation alarm (PK03-10).
5. Upper or lower detector deviation alarm (PK03-10).
6. Possible positive or negative rate bistable trip.
7. Possible low set point high level bistable trip.
8. Possible high set point high level bistable trip.

AUTOMATIC ACTIONS

1. Possible reactor trip.
2. Possible rod withdrawal block.
3. Possible loss or actuation of P-10.
4. Possible loss or actuation of P-8.
5. Possible turbine runback.

OBJECTIVES

1. Prevent undetectable reactivity or power changes.

IMMEDIATE OPERATOR ACTIONS

ACTION

1. If a channel fails high, place the rod control in manual and match Tavg and Tref, and stabilize the plant.

COMMENTS

1. If rods are used to match Ravg & Tref, place the rod stop bypass switch to the failed channel position to unblock the rods.

SUBSEQUENT OPERATOR ACTIONS

1. If one channel has failed during modes 1 and 2:

   a. Startup and power operation may continue provided action state #2 of Technical Specification 3.3.1 is complied with.

   b. At the Miscellaneous Indication Panel perform the following steps:

      1) Place the rod stop bypass switch to the failed channel position if not already done.

      2) Place the power mismatch bypass switch to the failed channel position.

      3) Place the detector current comparator upper and lower section switches to the failed channel position.

   c. At the Comparator and Rate Panel place the Defeat switch to the failed channel position.

   d. Within one hour:

      1) At the power range channel drawer remove the control and instrument power fuses.

      2) Refer to Table III A-1 of Volume 9 of the Plant Manual and place the associated bystables, for the failed channel in the tripped position.

2. If 2 or more channels are determined to be inoperable, follow the requirements of Tech. Spec. 3.0.3.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this event a significant event, as a minimum within one hour notify the NRC Bethesda Operations Center using the red phone from the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. If the event results in entry to a Technical Specification action statement requiring shutdown, designate the event a Notification of Unusual Event. Notify plant staff and response organizations in accordance with Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations "in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
MALFUNCTION OF THE RHR SYSTEM

APPROVED: O C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the steps to be taken following the failure of the residual heat removal loop while in operation.

SYMPTOMS

Any one or more of the following symptoms may be indicative of this emergency condition.

1. Reduced or zero flow as indicated on FI-970 A & B and FI-971 A & B and alarmed (PK02-16).
2. Increasing reactor coolant temperature.
3. Lack of temperature differential across the residual heat removal exchangers.
4. Residual heat removal pumps on-off lights, monitor box C lights and alarm (PK02-17).

AUTOMATIC ACTION

None

OBJECTIVES

1. Restore residual heat removal system to operation.
2. Initiate alternate means of heat removal if the RHR system cannot be restored to service.

IMMEDIATE OPERATOR ACTIONS COMMENTS

1. Verify that an RHR pump is in operation. If not, investigate reason and restore to operation if possible.
2. Verify that valves have not been inadvertently closed in the system.

COMMENTS

3. Verify that the failure is in the residual heat removal loop and not a loss of component cooling water to the heat exchanger. If component cooling water is lost, restore if possible.

SUBSEQUENT OPERATOR ACTIONS

1. If the residual heat removal loop cannot be restored to operation, isolate the RHR system from the reactor coolant system and start alternate means of heat removal as outlined.

   a. If the failure occurs when the reactor coolant system is being cooled below 350°F with the reactor head in place.

      1) Repressurize coolant system with charging pumps to minimum requirements for reactor coolant pump operation.

      2) Start reactor coolant pump.

      3) Start heat removal by steam dump from steam generators.

   b. If failure occurs when the reactor head is removed:

      1) Fill reactor cavity with water.

      2) Operate containment fans and coolers to help reduce the temperature.

2. Restore the residual heat removal system to an operable condition and return the system to normal operation.

3. Notify appropriate plant supervision.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent and Supervisor of Operations or their designated alternates.

   b. Designate this event a Significant Event. As a minimum, within one hour notify the NRC Operations Center using the red phone in the control room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to the NRC)" for a suggested format for this report. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. If a residual heat removal loop cannot be restored to operation (per step 1, page 2) designate this event an Alert. Notify plant staff and response organizations by implementing Emergency Procedure G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation.

DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

TITLE: EMERGENCY OPERATING PROCEDURE
LOSS OF A COOLANT LOOP RTD

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure outlines the steps to be taken following the failure of a primary coolant loop resistance temperature detector (RTD). The detector may be located on either the hot leg or cold leg and may fail either high (open circuit) or low (short circuit).

SYMPTOMS

1. ΔT deviation alarm (PK04-01).
2. Tavg deviation alarm (PK04-02).
3. Possible unwarranted rod motion if the rods are in automatic.
4. Auctioneered Tavg high alarm (PK04-10).
5. Tavg deviation from Tref alarm (PK04-03).
6. Possible rod insertion limit alarm.

AUTOMATIC ACTIONS

1. Possible rod motion in the decrease power direction (step in).

OBJECTIVES

1. Determine which loop contains the affected RTD and isolate the input from that loop to the automatic rod control system.
2. Verify proper control rod positioning by the automatic rod control system with the affected loop isolated, restore the affected loop to an operable condition and return the automatic rod control system to four loop operation.

IMMEDIATE OPERATOR ACTION

ACTION	COMMENTS
1. Determine whether an RTD failure has occurred by observing the ΔT and Tavg indicators for each loop.	

ACTION | COMMENTS

2. If improper rod motion has occurred, place the automatic control rod drive system on manual and return the rods to their correct positions.

3. Determine which RTD has failed and in which direction as follows:

   a. T_{avg} high and ΔT high means that a hot leg RTD failed high.

   b. T_{avg} high and ΔT low means that a cold leg RTD failed high.

   . T_{avg} low and ΔT high means that a cold leg RTD failed low.

   d. Tavg low and ΔT low means that a hot leg RTD failed low.

4. Defeat the Tavg and ΔT signals for the loop with the failed RTD by turning the defeat switches (T/411B and T/412B) from the normal position to the affected loop position.

5. Transfer the ΔT protection recorder to a different loop if necessary using selector switch T/411A.

SUBSEQUENT OPERATOR ACTIONS

ACTION | COMMENTS

1. Return the automatic rod control system to the automatic mode and check for proper operation.

2. Place the protection system bistables for the affected loop in the trip position.

FUNCTION	LOOP 1	LOOP 2	LOOP 3	LOOP 4
Overtemperature ΔT reactor trip	411C	421C	431C	441C
Overtemperature ΔT turbine runback (C-3)	411D	421D	431D	441D
Low T_{avg} feedwater isolation	412G	422G	432G	442G
Low-Low T_{avg}, P-12, SI, steam line isol	412D	422D	432D	442D
Overpower ΔT reactor trip	411G	421G	431G	441G
Overpower ΔT turbine runback (C-4)	411H	421H	431H	441H

ACTION	COMMENTS
3. Notify the Instrumentation and Control Supervisor so that the spare RTD may be quickly connected.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this event a Significant Event. As a minimum, within one hour notify the NRC Bethesda Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. Subsequent plant conditions should be reviewed against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
EMERGENCY SHUTDOWN

APPROVED: C T ... 2/15/82
PLANT MANAGER DATE

SCOPE

This procedure covers placing the reactor in the safe shutdown condition following an event (such as fire) that renders the normal emergency shutdown systems and equipment inoperable. Since the exact circumstances of the incident cannot be predicted ahead of time, this procedure is rather general and suggests different ways and means of accomplishing the goal. It is not considered that a LOCA or other major accident occurs simultaneously.

SYMPTOMS

The symptoms for an accident of this type could be practically unlimited. They will depend on the initiating event and will include such things as loss of indication on operating or standby equipment, annunciation of spurious conditions and possible overcurrent or other relay actions.

AUTOMATIC ACTIONS

The automatic actions which take place will depend on the source and magnitude of the initiating event. The could, however, include such things as:

1. Turbine-Generator trip
2. Reactor trip
3. Diesel generator start
4. Transfer to startup power
5. Actuation of sprinkler or deluge system.
6. Actuation of Cardox system.
7. Control room ventilation transfer

OBJECTIVES

1. To assure the reactor is shutdown (tripped) and is borated to the cold Xe free condition.
2. To establish a heat sink for the reactor.

IMMEDIATE OPERATOR ACTIONS

ACTIONS | COMMENTS

1. Regardless of whether the reactor has tripped, take the following actions:

   a. Determine the cause or source of the incident (fire, water, high energy line break, etc.).

   b. Take all possible actions to limit the consequences of the incident. This may include such things as deenergizing equipment, load centers or buses, securing equipment or lines and draining tanks.

   c. Determine the extent of the damage and the consequences of this damage in terms of equipment operability.

   d. Make an assessment of the potential damage and consequences if the situation cannot be readily controlled.

   e. If the possibility exists that redundant safeguards or shutdown systems may be affected or if they are affected, trip the reactor.

SUBSEQUENT OPERATOR ACTIONS

Once it has been determined that the redundant normal shutdown and cooldown systems are inoperable, the following general actions should be taken. First and most important is to get the reactor shutdown. Cooldown should then be initiated by any means possible. It is essential to get water into the steam generators for a heat sink and to insure natural circulation of the primary coolant.

ACTIONS

2. Shutdown the reactor.

   a. Trip the reactor in the normal manner from the control room.

   b. Check all rods inserted and reactor power decreasing rapidly.

   c. Shutdown both control rod drive MG sets.

   d. Check both reactor trip breakers and both bypass breakers open and rack them out.

3. Borate the reactor to the cold xenon free shutdown condition.

   It is highly desirable to get the reactor borated as soon as possible since conditions may deteriorate to where the heat tracing system is ineffective and boration becomes difficult to accomplish. Any of the following boration paths may be utilized as appropriate or necessary:

   a. Normal boration through the blender to the suction of the charging pumps.

   b. Emergency boration - refer to Emergency Operating Procedure OP-6 for details and various alternate flow paths.

   c. Boration through the normal charging path with the charging pump taking suction from the RWST.

      1) Open the charging pump suction valves from the RWST (8805A & B).

      2) Close the normal suction valves from the VCT (LCV-112B & C).

      3) If suction from the RWST cannot be achieved by gravity, it is possible to obtain suction from the RWST by running an RHR pump and opening valve 8804A.

COMMENTS

c. If power is not available, these valves can be operated manually.

ACTIONS	COMMENTS
d. Boration from the BIT with the charging pump taking suction from the RWST.	
1) Proceed as in Step 3 above.	
2) Open the BIT inlet valves (8803A & B).	
3) Open the BIT outlet valves (8801A & B).	
e. Boration with the charging pump taking suction directly from a boric acid storage tank.	e. Used as last resort due to RCP seal consideration.
1) This assumes that no boric transfer pump is available.	
2) Open the emergency borate valve (8104).	
3) Check open all valves in the flow path from the boric acid tanks to the emergency borate valve.	
4) Close the VCT outlet valves (LCV-112B&C).	
NOTE: This is necessary in order to reduce the suction header pressure to a point where gravity flow from the boric acid storage tanks is possible.	
5) Reduce charging pump flow to match the maximum suction flow and not cavitate the pumps.	

All of the above boration methods require at least one charging pump to be in service. If all charging pumps are inoperable, maintain the reactor in the hot condition. Do not attempt to reduce reactor pressure to the point where the safety injection pumps are usable at this time.

COMMENTS

4. Establish a Heat Removal Mechanism from the Steam Generators

   a. If condenser steam dump is available it should be used.

      Comment a. Handjacks on 40% steam dumps can be used if normal control is unavailable.

   b. With the reactor and turbine tripped the 35% atmospheric dump valves are unavailable.

   c. The 10% atmospheric dump valves should be available and should be used as the heat removal mechanism.

   d. If instrument air is lost, the 10% valves will control off back up N2. If back up N2 is lost, control the valve operation using back up air via toggle switch on VB-3.

   e. Cooldown capability is severely limited with only the safety valves available for heat removal and if 10% steam dumps can't be opened using Step 4 then the 10% steam dump valves should be manually opened using the handjack to provide a controlled heat removal mechanism.

5. Establish a Feedwater Supply to the Steam Generators.

   a. If the motor driven auxiliary feedwater pumps are operable, use them to feed the generators. If necessary, use the handjack to throttle the feedwater flow control valves. Valves are spring open, EH oil pressure to close.

   b. If the motor driven pumps are inoperable, use the turbine driven pump. Its flow control valves can be operated manually if necessary.

COMMENTS

1. If the speed control circuit for the turbine is inoperable, the speed can be controlled manually at the turbine governor.

2. The motor operated steam supply valves to the turbine can be opened manually if necessary.

3. If the turbine speed control system fails completely, the governor valve can be blocked open using the exposed linkage and speed can be manually controlled by manipulating the throttle stop valve manually.

3. It may be necessary to disconnect the governor valve linkage prior to blocking the valve open.

4. The turbine is designed to operate with steam supply pressure as low as 100 psig, but with reduced power output. The steam generator level should be built up to 50-60% while steam pressure is high so that the reduced flow rates at lower steam pressure can be tolerated during the cool-down.

5. The turbine driven pump can be utilized to cool the plant all the way down to the point where the RHR system can be placed in service.

6. With AFW pumps available cooldown can be accomplished provided there is suffient water available to the AFW pumps.

   a. If the condensate tank is lost, the fire tank can be lined up for suction. The suction valves can be manually opened if required.

   b. The raw water reservoir can be lined up to the AFW pumps via the make up water header.

COMMENTS

c. If the reservoir line is lost, line up the emergency pump at the reservoir and use the emergency hose to the condensate tank or M. U. Water header.

7. If condenser vacuum has been maintained, it is possible to use the main feedwater pumps in conjunction with the condensate/booster pumps to supply the steam generators.

   Comment 7. It may be necessary to reset the feedwater isolation signal.

   a. A condensate/booster pump set is required to supply suction to the main feed pump.

   b. Normal AC power must be available to the turbine high pressure lube oil pumps in order to latch the turbine and open the valves.

   c. Either the hand speed changer or the governor valve positioner may be used to control turbine speed.

   d. With the main unit tripped, only the high pressure steam supply will be available. The LP governor valves will be wide open before the HP governor valve starts to open.

   e. In the event of an extreme emergency with the condenser unavailable, block the low vacuum trip, break the casing rupture disk and attempt to run the turbine with exhaust to atmosphere.

   f. Once the steam generator pressure has been reduced to less than 600 psig, the condensate/booster pump sets can be used to feed water directly by pumping through the idle feed pumps. Using the feed water regulating valves for control.

COMMENTS

B. Primary System Cooldown Considerations

It is essential to maintain circulation of the reactor coolant to remove decay heat. If a reactor coolant pump is operable, it should be kept running. However, if all the coolant pumps are lost it is essential to maintain natural circulation. Verify natural circulation per Emergency Procedure OP-23.

NOTE: Natural circulation is caused by providing heat addition in one part of the system (decay heat from the core) and heat removal in another part of the system (steam removal from the generators). Natural circulation is better just after a trip when the decay heat is greater and it can be improved by increasing the heat removal rate. Natural circulation can be prevented by formation of a steam bubble in the tubes or by loss of heat removal capability. Primary system pressure and temperature should be monitored to insure that saturation conditions are not reached. Also circulation may stop if the steam generator secondary side temperature exceeds the primary side temperature.

a. In part B above the various boration methods were discussed. All of these methods require that a charging pump be operable. If all three charging pumps are inoperable, the goal becomes to get the primary system pressure reduced to where a safety injection pump can be used for boration (less than 1500 psig).

b. Check all pressurizer heaters deenergized.

c. Without any reactor coolant pumps or charging pumps there is no pressurizer spray available so pressure must be reduced by other means.

COMMENTS

d. Pressure can be reduced due to shrinkage caused by cooling the primary system. Reducing the primary system temperature to less than 510°F will produce a pressure of less than 1500 psig and permit boration by the safety injection pumps taking suction from the RWST.

Comment d. If all control rods inserted (or even with one stuck out) the amount of reactivity added by this temperature decrease will not be sufficient to make the reactor critical.

e. If it appears that the charging pumps may become inoperable, it may be necessary to blow down the primary system in order to get the pressure down to 1500 psig as quickly as possible.

Comment e. This method is used because the PORV's have no throttle capability.

1) Select one power relief valve for use and close its motor operated stop valve.

2) Open the power relief valve.

3) Initiate blowdown slowly by opening the stop valve until flow is detected then place the control switch in the STOP position.

4) Once flow has stabilized and if the PRT pressure and temperature are in control the stop valve can be opened further in small increments.

5) When pressure has been reduced to less than 1500 psig, close the relief valve and the power operated relief valve.

Comment 5) Verify closure by position indication lights and discharge pipe temperature elements.

f. Start a safety injection pump with suction from the RWST. Borate to the cold, xenon free condition.

g. If boration is slow, continue to cool down until the accumulator pressure is reached. Allow one accumulator at a time to discharge into the system to assist in boration.

COMMENTS

9. RHR System Operation.

   a. When primary system pressure and temperature have been reduced to 390 psig and 350°F place the RHR system in service to remove heat from the primary system.

   b. If the RHR system is not available or if its required support systems (component cooling and auxiliary salt water) are not available, proceed as follows.

      1) Continue to use the steam dump system to reduce RCS temperature down to 212°F.

      2) Continue to add feedwater to the steam generators.

      3) In order to cool down below 212°F, some kind of liquid release must occur since boiling will not occur.

      4) Open the blowdown valves on each steam generator all the way. The blowdown stream will provide a liquid heat removal path.

      5) If the blowdown flow is not large enough to provide adequate heat removal, increase the feedwater flow and start filling the steam generators.

      6) The addition of this cold water has a heat removal effect.

      7) Continue filling until water enters the main steam line. Line up all drains to bypass the traps and go directly to the condenser. This will provide an additional flow path.

a. The following steps also apply if the initiating event occurred with the RHR system in service.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. A more precise designation of this event will be determined by the cause and extent of damage. Refer to Emergency Procedure G-1 "Emergency Accident Classification and Plan Activation" for further classification guidance. As a minimum designate this event an Alert. Notify plant staff and response organizations required for this classification by implementing Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1.

TITLE: EMERGENCY OPERATING PROCEDURE
TANK RUPTURES

APPROVED: R C Thornberry 2/15/82
PLANT MANAGER DATE

SCOPE

This procedure outlines the steps to take in the event a gas decay tank, liquid holdup tank or volume control tank ruptures and releases radioactive gas and/or liquid to the auxiliary building.

SYMPTOMS

1. Plant vent monitor high radiation alarm and containment ventilation isolation.

2. Possible LHUT or VCT low level alarm.

3. Gas decay tank, LHUT, or VCT low pressure alarm.

4. Persons near the affected areas may find themselves contaminated when checking out at access control or exposed to above normal radiation levels.

AUTOMATIC ACTIONS

1. High radiation on plant vent air particulate monitors (R-28 A or B) or plant vent radio gas monitors (R-14 A or B) initiates containment ventilation isolation.

2. At 5% VCT level charging pump suction valves from RWST 8805 A & B open and VCT outlet valves LCV's 112 B & C close.

3. Low pressure on LHUT trips LHUT recirculation pump.

OBJECTIVES

1. Alert on site personnel.

2. Evaluate the release and take appropriate protective measure.

IMMEDIATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Initiate the site emergency signal	1. Evacuation of personnel from affected area

ACTIONS

2. Place auxiliary building ventilation in charcoal filter mode by SI test signal at POV 2.

3. Either shutdown or place the unaffected units auxiliary building ventilation system in the charcoal filter mode of operation.

COMMENTS

2. To reduce Iodine release from plant vent.

SUBSEQUENT OPERATOR ACTIONS

ACTIONS

1. Evacuate all personnel from the affected area.

2. Refer to the following emergency operating procedures applicable:

   R-1 Personnel Injury (Radiological Related) and/or overexposure

   R-2 Release of Airborne Radioactive Materials

   R-4 High External Radiation

   R-5 Radioactive Liquid Spill

3. Isolate the release

   a. For a gas decay tank rupture:

      1) Select the affected tank to standby so that it is neither filling nor providing recycle gas.

   b. For a LHUT rupture:

      1) Stop any transfer or recirculation operation involving the affected LHUT.

      2) Line up a different LHUT to receive letdown from the primary system other than the affected LHUT.

ACTIONS	COMMENTS
3) Stop any cover gas recycle to the affected LHUT.	
4) Check VCT and accumulator test line discharge lined up to another LHUT and isolate discharge to affected LHUT.	
c. For a VCT rupture	
1) Place the VCT level control LCV-112A in the DIVERT TO HOLDUP TANK position.	
2) Check transferred or transfer charging pump suction from VCT to RWST (open 8805 A&B and close LCVs 112 B&C).	
3) Terminate VCT makeup.	
4) Secure hydrogen supply to the affected unit's VCT at the hydrogen bottle rack.	4) Make arrangements for an alternate source of hydrogen makeup to Unit 1 generator.
5) Check closed or close VCT to vent header stop valve 8101.	
6) Commence a controlled reactor shutdown.	6) Drop load on unit as necessary to maintain rod position and Tavg approximately equal to Tref.
4. Verify containment ventilation isolation and reset containment ventilation isolation trains A & B.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. The precise designation of this event will be determined by the radiological effect of the leak. Refer to Emergency Procedure R-2 "Release of Airborne Radioactive Material" and R-4 "High Radiation (In-plant). As a minimum, in the absence of data on radiation levels or release rates, designate this event a Notification of Unusual Event. Notify plant staff and response organizations required by Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations in accordance with Emergency Procedure G-2 "Accident Classification and Emergency Plan Activation."

Pacific Gas and Electric Company

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-26

REVISION 1

DATE 2/12/82

TITLE: EMERGENCY OPERATING PROCEDURE
EXCESSIVE FEEDWATER FLOW

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the steps to take in the event of excessive feedwater flow to one steam generator resulting from the failure of a feedwater control valve in the open position. This malfunction causes a cold water transient that is terminated by either a steam generator high-high level turbine trip or by ahigh power reactor trip. It is also possible that the event may result in a Safety Injection Signal due to steam line differential pressure.

SYMPTOMS

1. Steam flow - feedwater flow mismatch alarm.
2. Steam generator high level alarm.
3. ΔT deviation alarm (PK04-1).

AUTOMATIC ACTIONS

1. Possible turbine trip on P-14 signal.
2. Possible reactor trip if above P-7.
3. Possible feedwater isolation on P-14 signal.
4. Possible main feedwater pumps trip on P-14 signal.
5. Possible S. I. due to steam line ΔP.

OBJECTIVES

1. To terminate cooldown without a reactor trip.
2. To reestablish S/G level control.

IMMEDIATE OPERATOR ACTIONS	COMMENTS
1. Place the feedwater valve control on MANUAL and attempt to throttle the valve.	1. Attempt to stabilize the steam generator level.
2. If the valve cannot be throttled, close the motor operated feedwater isolation valve for the affected steam generator.	
3. If feedwater for the affected steam generator cannot be isolated, trip the reactor and the turbine.	
4. Close the M.S. isolation valve, and the STM supply valve to the steam driven AFW pump if applicable, on the affected steam generator.	
5. Verify running or start the AFW pumps and throttle flow to the affected steam generator if required.	5. The affected steam generator may or may not have a high level at this time.
6. Verify the 10% atmosphere steam dump holding 1035 psig on the affected steam generator.	
SUBSEQUENT OPERATOR ACTIONS	
1. If a reactor trip has not occurred and the steam generator level increase has halted.	
a. Adjust the steam generator level controller in manual to maintain the level at the L_{ref} value.	a. Steam generator setpoint indicator is on VC-3.
b. Determine if the event was due to a malfunction of a steam flow or feed-water flow control channel by selecting the alternate channel.	
c. If a flow control channel has mal-functioned, use the alternate flow channel and return the level control to automatic.	c. Before placing the level control back into auto-matic, match actual level to reference level and feed flow to steam flow.
2. If the reactor trips, refer to Emergency Operating Procedure No. OP-5.	
3. If a Safety Injection signal is initiated refer to Emergency Operating Procedure No. OP-0.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

    a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

    b. Designate this a significant event. As a minimum, within one hour notify the NRC Bethesda Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

    c. Review subsequent plant conditions against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
IRRADIATED FUEL DAMAGE

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure gives guidance in the event that an irradiated fuel element is damaged to the extent that fission products are released. The element may be damaged in either the containment during core alterations, or in the fuel handling building during fuel movements. In both cases, the irradiated fuel element is under water; and in both cases, operating personnel are working with the irradiated fuel element.

SYMPTOMS

1. Personnel observe the damage taking place or observe gas bubbles rising from the irradiated fuel element.

2. Hi containment area radiation alarm.

3. Hi containment radiogas and/or particulate alarm.

4. Hi fuel building area radiation monitor alarm.

AUTOMATIC ACTIONS

1. Containment Ventilation Isolation if fuel damage in containment.

2. Fuel Handling Building Ventilation System transfers to Iodine Removal mode of operation if fuel damage in Fuel Handling Building.

OBJECTIVES

1. Evacuate all personnel from the affected building.

2. Maintain containment integrity if the damage is in containment.

3. Ensure Iodine removal if damage is in the Fuel Handling Building.

IMMEDIATE OPERATOR ACTIONS

A. IRRADIATED FUEL DAMAGE IN THE FUEL HANDLING BUILDING.

ACTIONS	COMMENTS
1. The Senior Operator in charge of fuel handling should immediately evacuate all personnel from the building and	1. Be on the alert for possible fuel damage while working with irradiated fuel.

ACTIONS	COMMENTS
notify the Control Room of the problem.	Fuel damage is caused by dropping an element, allowing an element to strike another hard surface, imposing bending forces on elements or by dropping a heavy object onto a stored element. When fuel damage is known to have occurred by evidence of gas bubbles rising from an element or, in the opinion of the Senior Operator in charge, fuel damage may have occurred, Action 1 should be carried out.
2. If the Control Room Operator receives a Fuel Handling Building Area radiation hi level alarm or is notified by the Senior Operator in charge of fuel handling that damage to irradiated fuel has occurred, he will proceed as follows:	
a. Verify the Fuel Handling Building evacuation horn has sounded.	
b. Verify auto actuation or manually initiate Fuel Handling Building Ventilation System Iodine mode of operation.	

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Verify by head count and Radiation Area Entry Log that all personnel are out of the Fuel Handling Building.	

ACTIONS	COMMENTS
2. Verify Iodine Removal Mode of Operation is in service by checking the PØV. Ventilation Panel in the Control Room.	3. Dampers correct to place air flow thru charcoal filters and fans operating.
3. Monitor the stack gas radiation monitors to determine if any off-site release is evident.	4. Notify plant management of the problem promptly.

IMMEDIATE OPERATOR ACTIONS

B. IRRADIATED FUEL DAMAGE IN THE CONTAINMENT BUILDING

ACTIONS	COMMENTS
1. The Senior Operator in charge of fuel handling should immediately evacuate all personnel from the building and notify the Control Room of the problem.	1. Be on the alert for possible fuel damage while working with irradiated fuel. Fuel damage is caused by dropping an element, allowing an element to strike another hard surface, imposing bending forces on elements or by dropping a heavy object onto a stored element. When fuel damage is known to have occurred by evidence of gas bubbles rising from an element or, in the opinion of the Senior Operator in charge, fuel damage may have occurred, Action 1 should be carried out.
2. If the Control Room Operator receives a Containment Building Area Radiation Hi Level Alarm, hi radiogas or hi particulate activity alarm or is notified by the Senior Operator in charge of fuel handling that damage to irradiated fuel has occurred, he will proceed as follows: a. Verify or initiate the Containment Building evacuation horn. b. Verify auto actuation or manually initiate Containment isolation.	

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Verify by head count and Radiation Area Entry Log that all personnel are out of the containment.	
2. Verify Containment vent isolation by observing valve positions correct. (No monitor lights lit on Phase A Postage Stamp Monitor Light box.)	
3. Monitor the stack gas radiation monitors to determine if any offsite release is evident.	3. Notify plant management of the problem promptly
4. Start all available containment iodine removal units.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. In the absence of data on in-plant radiation levels or radiation release rate, designate this event an Alert. Notify plant staff and response organizations by implementing Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

   b. If a release is evident, or in-plant radiation levels are high, refer to Emergency Procedure R-2 "Release of Airborne Radioactive Materials" and R-4 "High Radiation (In-plant)" for further classification guidance.

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-28
REVISION 1
DATE 2/12/82
PAGE 1 OF 3

TITLE: EMERGENCY OPERATING PROCEDURE
STARTUP OF AN INACTIVE REACTOR COOLANT LOOP

APPROVED: R C T [signature] 2/15/82
PLANT MANAGER DATE

SCOPE

The purpose of this procedure is to provide instruction in the event of a rapid reactivity addition due to the inadvertent starting of an idle reactor coolant pump with reactor power greater than 25%.

If the reactor is operated at power with one pump out of service, the idle loop not only experiences reverse flow but also a temperature drop across the steam generator hence, the temperature in the idle hot leg section of pipe will be colder than the core inlet temperature (this temperature being a function of load). When the idle pump is started, a reactivity excursion will result from the colder water being introduced into the core (EOL being the worst case).

SYMPTOMS

1. Control rods move in if in auto.

2. NIS instrumentation indicate increasing neutron flux level.

3. Power range high neutron flux rod stop alarm.

AUTOMATIC ACTIONS

1. The control rods will drive in at high speed.

2. Possible reactor trip due to:

    a) Power range high neutron flux or

    b) Turbine trip from P-14 signal (idle steam generator water level will jump rapidly whe hotter reactor coolant water is introduced).

OBJECTIVES

1. Stabilize the Reactor Coolant System.

IMMEDIATE OPERATOR ACTIONS

ACTIONS — COMMENTS

1. If reactor trip did not occur, check for violation of rod insertion limits.

   a. Emergency borate if necessary to return the rods out past the insertion limit.

2. If the reactor trips, go to OP-5

SUBSEQUENT OPERATOR ACTIONS — COMMENTS

ACTIONS

1. After the power excursion:

   a. Check failed fuel monitor reading.

   b. Take coolant chemistry samples for specific activity.

   c. If activity is higher than acceptable limits, refer to Subsequent Actions of Emergency Operating Procedure No. 14.

2. If a reactor trip has occurred, maintain hot standby conditions until authorized to proceed by the Plant Superintendent.

3. If a reactor trip did not occur, stabilize the unit and notify the Plant Superintendent and Supervisor of Operations.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations, and Plant Manager or their designated alternates.

   b. Designate this event a significant event and as a minimum, within one hour notify the NRC Bethesda Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. Review subsequent plant conditions against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
EXCESSIVE LOAD INCREASE

APPROVED: R C Thornberry 2/15/82
PLANT MANAGER DATE

SCOPE

The purpose of this procedure is to provide instruction and guidance in the event of an excessive load increase. An excessive load increase event is defined as a rapid increase in steam flow such that a power mismatch exists between steam generator load demand and reactor core power. The event could be the result of a failure in turbine control or steam dump control.

SYMPTOMS

1. Many symptoms vary depending on whether the rods are in manual or auto or if the event occurs at BOL or EOL. However, two indications behave consistently.

   a. ΔT increases significantly.

   b. Neutron flux levels increase significantly.

2. TAVG-TREF deviation alarm due to TAVG < TREF.

3. Possible generator load increasing.

AUTOMATIC ACTIONS

1. Possible rod stop and turbine run back on OTΔT or OPΔT. (C-3 and C-4).

2. Possible rod stop on high neutron flux (C-1 or C-2).

3. Possible reactor trip on OTΔT or OPΔT.

4. Possible reactor trip on high neutron flux (IR, PR low setpoint or PR high setpoint).

OBJECTIVES

1. To terminate the excessive load increase.

2. To match TREF and TAVG.

IMMEDIATE OPERATOR ACTIONS	COMMENTS
1. If T_{REF} increase is unexpected or is increasing at an abnormally high rate:	
a. Depress the HOLD pushbutton on the turbine control console.	a. Pushbutton should be backlighted.
1) Check the selected LOAD rate value if a load change is in progress.	1) Maximum Administrative Limit is 50MW/min.
2) Match T_{REF} to T_{AVG}.	2) Decrease turbine load to lower T_{REF}.
b. If the load increase failed to halt when the HOLD pushbutton was depressed, press the TURBINE MANUAL pushbutton to establish turbine control.	b. Pushbutton should backlight.
c. If the turbine load continues to increase trip the turbine.	c. If above P-7 the turbine trip will initiate a reactor trip, go to OP-5. If below P-7, the reactor will be maintained on steam dump.
2. If T_{REF} is not increasing but ΔT and neutron flux levels are increasing, check the steam dump valve positions.	2. Postage stamp lights on VB-3
a. If dump valves are open and steam dump is not warranted, place the STEAM DUMP Train A and Train B selector switches to the OFF position and the valves should close.	a. Check C-7A, C-7B, C-8 and if $T_{AVG} > T_{REF}$. The dump valves may have been armed previously but never reset.
b. If a 10% steam generator relief valve is open and steam dump is not warranted, place the control in manual and close the valve.	

SUBSEQUENT OPERATOR ACTIONS

COMMENTS

1. If an erroneous LOAD RATE value was selected for the turbine load increase, select an appropriate value and operation may continue.

2. If the event was due to a turbine control failure leave the turbine on manual and do not increase turbine load unless authorized by the Plant Superintendent.

3. If the event was due to a steam dump valve opening, with the valve now closed, place the control rods and turbine control in automatic if appropriate. Do not increase turbine load unless authorized by the Plant Superintendent.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this event a significant event. As a minimum, within one hour notify the NRC Bethesda Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. Review subsequent plant conditions against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

Pacific Gas and Electric Company

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-31

REVISION 1

DATE 2/12/82

TITLE: EMERGENCY OPERATING PROCEDURE
SYSTEM UNDERFREQUENCY

APPROVED: [signature] 2/15/82
PLANT MANAGER DATE

SCOPE

This procedure outlines the steps to be taken following an immediate drop in system frequency due to excessive system load or a deficiency in system generating capacity.

SYMPTOMS

Any one or more of the following symptoms may be indicative of this emergency condition, depending on initial plant load and magnitude of the frequency drop.

1. Reduced frequency as indicated on generator frequency meters and recorder and reduced turbine speed as indicated on turbine speed RPM indicators.

2. Reactor coolant flow decreases with frequency.

3. Generator underfrequency alarm (PK14-05).

4. Possible condensate system transient bypass alarm (PK10-7) (if turbine trips).

5. Possible reactor trip, reactor coolant pumps trip, and unit trip.

AUTOMATIC ACTION

1. Generator separates from the system by underfrequency relay (58 Hz after 3 minutes, 57 Hz after 1 minute, or 55 Hz after 30 cycles) and carries station auxiliary load. Turbine control changes to speed control.

2. Possible reactor trip and reactor coolant pumps trip by RCP bus underfrequency (54 Hz 2/3 in 1/2 busses).

3. Possible unit trip with transfer of auxiliary power to start-up power or transfer of vital busses to diesel generators.

4. Possible condensate system transient bypass actuation.

5. Steam dump actuation.

OBJECTIVE

1. Maintain main unit turbine generator operating as a house unit at normal frequency and voltage.

2. Prevent turbine blading damage from harmonic vibrations caused by sustained operation at lowered frequency.

3. Establish stabilized reactor coolant system and steam generator conditions in hot standby mode following reactor trip.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. If reactor trips, follow emergency operating procedure OP-5.	
2. If generator trips, follow emergency operating procedure OP-34.	
3. If electrical power is lost, follow emergency operating procedure OP-4.	
4. If generator separates from system carrying auxiliary load, check or adjust frequency and voltage to normal as necessary.	

SUBSEQUENT OPERATOR ACTIONS

1. Refer to emergency operating procedures OP-5, OP-34, and/or OP-4.

2. Verify steam dump operation and load transient bypass operation as necessary.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this event a significant event. As a minimum, within one hour notify the NRC Bethesda Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. Review subsequent plant conditions against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

Pacific Gas and Electric Company

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-34
REVISION 1
DATE 2/12/82
PAGE 1 OF 4

TITLE: EMERGENCY OPERATING PROCEDURE
GENERATOR TRIP - FULL LOAD REJECTION

APPROVED: [signature] 2/15/82
PLANT MANAGER DATE

SCOPE

This procedure gives guidance in the event the main generator trips. In this case, the unit experiences a full load rejection but remains in service to supply its own electrical needs.

SYMPTOMS

1. Generator trip alarm.
2. Steam dump activated alarm.
3. Generator output reduces to approximately 5%.
4. Control rods stepping in at maximum rate.

AUTOMATIC ACTIONS

1. Both main generator PCB's open.
2. Steam dump valves open on TAVG-TREF program.
3. Control rods respond to TAVG-TREF mismatch by stepping in at maximum rate.
4. Turbine governor controls turbine speed at 1800 RPM.
5. Condensate system transient bypass activates.

OBJECTIVES

1. To prevent a reactor trip.
2. To provide the reactor with a heat sink.
3. To prevent a turbine trip.
4. To stabilize the plant.

IMMEDIATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. If a reactor trip occurs, go to the reactor trip procedure OP-5.	

ACTIONS	COMMENTS
2. Verify steam dump valves open and T_{avg} decreasing towards T_{ref}.	2. T_{ref} should immediately decrease to about 550°F.
3. Verify control rods in auto and stepping in at maximum rate and nuclear power decreasing.	3. Maximum rate is 72 steps per minute.
4. Verify turbine speed being held at 1800 RPM by the governor.	4. 60 cycles on the cycle meter.

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Verify steam dump valves closing as T_{avg} decrease.	
2. Verify PZR pressure remains within the pressure control band of 2200 psig to 2275 psig.	2. PZR pressure will first increase then decrease and finally remain constant at about 2235 psig.
3. Verify PZR level decreasing.	3. The PZR pressure and level should trend with each other. PZR level should stabilize at about 25% for 10% load.
4. Verify secondary steam pressure increasing.	
5. At 10% reactor power,place the control rods on manual control to prevent further rod insertion.	5. At 10%, only the 40% cooldown steam dump valves should be open.
6. Shutdown one main feedwater pump, if possible leave the other feedwater pump on auto; however be prepared to transfer to manual if conditions warrant.	
7. Adjust the control rod positions to insure a continuous steam dump to the condenser. The amount of steam dump should be the minimum required to allow the remaining feedwater pump and steam generator level controls to remain on automatic.	

ACTIONS	COMMENTS
8. The unit should now be stable at an electrical load sufficient to supply its own auxiliaries. Control rods are on manual with the operator controlling T_{avg} and steam dump.	
9. Reset the load transient bypass feature.	
10. If the unit is to remain operating in this condition for any length of time, perform the following:	
a. Shutdown one condensate and booster pump set.	
b. Shutdown the No. 2 heater drain pump.	
c. Shutdown other auxiliary equipment as needed.	
d. Continue to operate with the turbine in operator automatic under the control of the governor.	d. The governor will hold the unit at 1800 RPM.
e. Leave the voltage regulator in automatic and adjust voltage to the desired value of 25KV using the voltage adjuster.	e. In this mode of operation, the operator has no control over VARS or power factor.
11. Notify appropriate Plant Management.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this event a significant event. As a minimum, within one hour notify the NRC Bethesda Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. Review subsequent plant conditions against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
TURBINE TRIP

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure provides guidance in the event of a turbine trip. If the unit is above P-7 (10% power), the turbine trip will cause a reactor trip; if below P-7 the reactor will not trip and will use steam dump for its heat sink. With the unit above P-7, the operator will use the reactor trip procedure OP-5 to handle both the reactor trip and turbine trip; below P-7 this procedure will be used.

SYMPTOMS

1. Turbine trip alarm.
2. Steam dump alarm.

AUTOMATIC ACTIONS

1. Steam dump maintains heat sink.
2. Generator trips after 30 seconds.

OBJECTIVES

1. To provide a heat sink for the reactor.
2. To stabilize the plant.

IMMEDIATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. If the reactor trips, go immediately to reactor trip procedure OP-5.	
2. Verify steam dump holding secondary steam pressure at the pressure control setpoint.	2. Steam dump will be in the steam pressure mode.
3. Verify all 4 turbine stop valves closed.	

ACTIONS

4. erify generator trips after 30 second delay.

COMMENTS

4. The auxiliaries will transfer to SU power if they are being supplied by the main generator.

SUBSEQUENT OPERATOR ACTIONS

ACTIONS

1. Verify auxiliary feedwater pumps running and delivering flow to the steam generators.

2. Start the steam driven auxiliary feedwater pump if required.

3. Verify steam generator auxiliary feedwater valves holding 33% steam generator level.

4. Verify turbine speed decreasing below 1800 RPM.

5. Reduce reactor power to 3% and maintain stable conditions.

6. Verify condenser vacuum holding at the pre-event value.

7. Investigate the cause of the turbine trip. If the turbine tripped due to controls being on hand, adjust all plant parameters to within limits and proceed with a normal startup.

8. If the turbine tripped due to equipment problems, proceed to hot standby or cold shutdown to effect repairs.

9. If the turbine is to be put on turning gear, verify the following:

COMMENTS

7. The most likely case is steam generator level controls on hand.

TITLE: TURBINE TRIP

ACTIONS	COMMENTS
a. AC bearing oil pump and Hi pressure seal oil backup pump running after oil pressure decays to 11 psig.	
b. Turbine bearing lift pump starts at 600 RPM turbine speed.	
c. Main and feedpump turbines on turning gear at 0 RPM.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Operations and Plant Manager or their designated alternates.

   b. Designate this event a significant event. As a minimum, within one hour notify the NRC Bethesda Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. Review subsequent plant conditions against the classification criteria in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

PG&E **Pacific Gas and Electric Company**

NUMBER EP OP-37
REVISION 1
DATE 2/12/82

DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

TITLE: EMERGENCY OPERATING PROCEDURE
LOSS OF PROTECTION SYSTEM CHANNEL

APPROVED: R C Thornberry 2/13/82
PLANT MANAGER DATE

SCOPE

This procedure provides general guidelines for operator actions if a protection system instrument channel fails. A "Protection System Instrument Channel" is any instrument or instrument loop which has an input to the Solid State Protection System. It assumes that any abnormal indication or control transient has been positively identified as a result of instrument failure and not as a result of an actual emergency situation.

SYMPTOMS

1. Out of Limits Alarm.

2. One indicator reading significantly higher (or lower) than other indicators of the same parameter (disagreement between redundant indicators).

3. Possible control system aberration.

OBJECTIVES

1. Stabilize plant.

2. Put Protection System Logic in a safe configuration.

IMMEDIATE OPERATOR ACTIONS

ACTION	COMMENTS
1. Separate the failed channel from any associated control loop.	1. May require manual control of the affected parameter.

SUBSEQUENT OPERATOR ACTIONS

ACTION	COMMENTS
1. Bring affected parameter into normal operation range.	
2. Select a redundant channel to replace failed channel in control loops or defeat failed channel input to control loops.	
3. Return parameter control to AUTO if possible.	

#	ACTION	COMMENTS
4.	Trip bistables associated with failed instrument as noted in Table IIIA-1 (Protection input bistables to be tripped in the event of instrument channel failure) in Volume 9 of the Plant Manual.	4. Appendix 1 attached is for information only. Consult Volume 9 for the most current information.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Notify the Plant Superintendent, Supervisor of Plant Operations and the Plant Manager.

   b. Designate this a significant event. As a minimum, within one hour notify the NRC Operations Center using the red phone in the Control Room. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10 CFR Part 50.72, (Notification of Significant Events).

   c. If the Loss of Protection System Channel requires entering a technical specification action statement requiring plant shutdown, designate this event a <u>Notification of Unusual Event</u>. Notify plant staff and response organizations by implementing Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

APPENDIX 1

LOSS OF PROTECTION SYSTEM CHANNEL

INSTRUMENT CHANNEL DESCRIPTION	CHANNEL NO. DESCRIP.	TECH SPECS ACTION NO.	PROTEC. CHANNEL	RACK NO.	BISTABLE SWITCH NO. BS -
Power Range Neutron Ø For level trips, and rate trips, pull Inst. & Control Power Fuses on Power Range Drawers. The Bistables to be tripped are for OTΔT and Rodstop only.	N41	N/A 2	I	2 2	411D (Rodstop) 411C (OTΔT)
	N42	N/A 2	II	6	421D (Rodstop) 421C (OTΔT)
	N43	N/A 2	III	13 13	431D (Rodstop) 431C (OTΔT)
	N44	N/A 2	IV	15 15	441D (Rodstop) 441C (OTΔT)
Intermediate Range Neutron Ø If channel is inoperable, bypass trip and Rodstop, comply with TS Action	N35&N36	3			None
Source Range Neutron Ø If channel is inoperable, bypass trip and comply with TS Action	N31&N32	4			
Either Loop No. 1 RTD TE 411A or B	OTΔT	2 or 9 N/A	I I	2 2	411C (Trip) 411D (Rodstop)
	OPΔT	6 or 9 N/A	I I	2 2	411G (Trip) 411H (Rodstop)
	Lo Tavg	N/A	I	2	412G
	FW Iso P-12	14 or 15	I	2	412D
Either Loop No. 2 RTD TE 421A or 421B	OTΔT	2 or 9 N/A	II II	6 6	421C (Trip) 421D (Rodstop)
	OPΔT	6 or 19 N/A	II II	6 6	421G (Trip) 421H (Rodstop)
	Lo Tavg	N/A	II	6	422G
	FW Iso P-12	14 or 15	II	6	422D

APPENDIX 1 (Cont'd)

INSTRUMENT CHANNEL DESCRIPTION	CHANNEL NO. DESCRIP.	TECH SPECS ACTION NO.	PROTEC. CHANNEL	RACK NO.	BISTABLE SWITCH NO. BS -
Either Loop No. 3 RTD TE 431A or 431B	OTΔT	2 or 9	III	13	431C (Trip)
		N/A	III	13	431D (Rodstop)
	OPΔT	6 or 9	III	13	431G (Trip)
		N/A	III	13	431H (Rodstop)
	Low Tavg FW Iso	N/A	III	13	432G
	P-12	14 or 15	III	13	432D
Either Loop No. 4 RTD TE 441A or 441B	OTΔT	2 or 9	IV	15	441C (Trip)
		N/A			441D (Rodstop)
		6 or 9	IV	15	441G (Trip)
		N/A	IV	15	441H (Rodstop)
	Lo Tavg FW Iso	N/A	IV	15	442G
	P-12	14 or 15	IV	15	442D
Pressurizer Pressure 455	OTΔT	2, 9	I	2	411C (Trip)
		N/A	I	2	411D (Rodstop)
	Hi P Trip	6	I	1	455A
	Lo P Trip	6	I	1	455C
	Lo P SI	14	I	1	455D
	P-11	N/A	I	1	455B
Pressurizer Pressure 456	OTΔT	2, 9	II	6	421C (Trip)
		N/A	II	6	421D (Rodstop)
	Hi P Trip	6	II	10	456A
	Lo P Trip	6	II	10	456C
	Lo P SI	14	II	10	456D
	P-11	N/A	II	10	456B
Pressurizer Pressure 457	OTΔT	2, 9	III	13	431C (Trip)
		N/A	III	13	431D (Rodstop)
	Hi P Trip	6	III	12	457A
	Lo P Trip	6	III	12	457C
	Lo P SI	14	III	12	457D
	P-11	N/A	III	12	457B

APPENDIX 1 (Cont'd)

INSTRUMENT CHANNEL DESCRIPTION	CHANNEL NO. DESCRIP.	TECH SPECS ACTION NO.	PROTEC. CHANNEL	RACK NO.	BISTABLE SWITCH NO. BS -
Pressurizer Pressure 474	OTΔT	2,9	IV	15	441C (Trip)
		N/A	IV	15	441D (Rodstop)
	Hi P Trip	6	IV	14	474C
	Lo P Trip	6	IV	14	474A
Pressurizer Level 459	Hi L Trip	7	I	1	459A
	Lo L SI	14	I	1	459B
Pressurizer Level 460	Hi L Trip	7	II	10	460A
	Lo L SI	14	II	10	460B
Pressurizer Level 461	Hi L Trip	7	III	12	461A
	Lo L SI	14	III	12	461B
RCL No. 1 Flow 414	Lo F Trip	7	I	1	414
415	Lo F Trip	7	II	10	415
416	Lo F Trip	7	III	12	416
RCL No. 2 Flow 424	Lo F Trip	7	I	1	424
425	Lo F Trip	7	II	10	425
426	Lo F Trip	7	III	12	426
RCL No. 3 Flow 434	Lo F Trip	7	I	1	434
435	Lo F Trip	7	II	10	435
436	Lo F Trip	7	III	12	436
RCL No. 4 Flow 444	Lo F Trip	7	I	1	444
445	Lo F Trip	7	II	10	445
446	Lo F Trip	7	III	12	446
Stm. Gen. No. 1 Water Level 517	Lo Lo Trip	7	IV	16	517B
	HiHiTurb "	14	IV	16	517A
	Lo Lvl Trip	7	IV	16	517C
518	Lo Lo Trip	7	III	11	518B
	HiHiTurb "	14	III	11	518A
	Lo Lvl Trip	7	III	11	518C
519	Lo Lo Trip	7	II	10	519B
	HiHiTurb "	14	II	10	519A

APPENDIX 1 (Cont'd)

INSTRUMENT CHANNEL DESCRIPTION		CHANNEL NO. DESCRIP.	TECH SPECS ACTION NO.	PROTEC. CHANNEL	RACK NO.	BISTABLE SWITCH NO. BS -
Stm. Gen. No. 2 Water Level	527	Lo Lo Trip	7	IV	16	527B
		HiHiTurb "	14	IV	16	527A
		Lo Lvl Trip	7	IV	16	527C
	528	Lo Lo Trip	7	III	11	528B
		HiHiTurb "	14	III	11	528A
		Lo Lvl Trip	7	III	11	528C
	529	Lo Lo Trip	7	I	1	529B
		HiHiTurb "	14	I	1	529A
Stm. Gen. No. 3 Water Level	537	Lo Lo Trip	7	IV	16	537B
		HiHiTurb "	14	IV	16	537A
		Lo Lvl Trip	7	IV	16	537C
	538	Lo Lo Trip	7	III	11	538B
		HiHiTurb "	14	III	11	538A
		Lo Lvl Trip	7	III	11	538C
	539	Lo Lo Trip	7	I	4	539B
		HiHiTurb "	14	I	4	539A
Stm. Gen. No. 4 Water Level	547	Lo Lo Trip	7	IV	16	547B
		HiHiTurb."	14	IV	16	547A
		Lo Lvl Trip	7	IV	16	547C
	548	Lo Lo Trip	7	III	11	548B
		HiHiTurb "	14	III	11	548A
		Lo Lvl Trip	7	III	11	548C
	549	Lo Lo Trip	7	II	8	549B
		HiHiTurb "	14	II	8	549A
Stm. Gen. No. 1 Stm. Flow	512	MismatchTrip	7	I	3	510B
		SI	14/15	I	3	512
	513	MismatchTrip	7	II	7	511B
		SI	14/15	II	7	513
Stm. Gen. No. 2 Stm. Flow	522	MismatchTrip	7	I	3	520B
		SI	14/15	I	3	522
	523	MismatchTrip	7	II	9	521B
		SI	14/15	II	9	523

APPENDIX 1 (Cont'd)

INSTRUMENT CHANNEL DESCRIPTION		CHANNEL NO. DESCRIP.	TECH SPECS ACTION NO.	PROTEC. CHANNEL	RACK NO.	BISTABLE SWITCH NO. BS -
Stm. Gen. No. 3 Stm. Flow	532	MismatchTrip	7	I	5	530B
		SI	14/15	I	5	532
	533	MismatchTrip	7	II	7	531B
		SI	14/15	II	7	533
Stm. Gen. No. 4 Stm. Flow	542	MismatchTrip	7	I	5	540B
		SI	14/15	I	5	542
	543	MismatchTrip	7	II	9	541B
		SI	14/15	II	9	543
Stm. Gen. No. 1 Feed Flow	510	MismatchTrip	7	I	3	510B
	511	MismatchTrip	7	II	7	511B
Stm. Gen. No. 2 Feed Flow	520	MismatchTrip	7	I	3	520B
	521	MismatchTrip	7	II	9	521B
Stm. Gen. No. 3 Feed Flow	530	MismatchTrip	7	I	5	530B
	531	MismatchTrip	7	III	7	531B
Stm. Gen. No. 4 Feed Flow	540	MismatchTrip	7	I	5	540B
	541	MismatchTrip	7	II	9	541B
Containment Pressure	934	SI	14	IV	16	934A
		Phase B	16	IV	16	934B
	935	SI	14	III	11	935A
		Phase B	16	III	11	935B
	936	SI	14	II	8	936A
		Phase B	16	II	8	936B
	937	Phase B	16	I	4	937B
Steam Line 1 Pressure	514	ΔPSI $P_1<P_2$	14/15	I	3	514A
		ΔPSI $P_2<P_1$	14/15	I	3	514B
		HiStmFlo SI	14/15	I	3	512
		MismatchTrip	7	I	3	510B
	515	ΔPSI $P_1<P_3$	14/15	II	7	515A
		$P_3<P_1$	14/15	II	7	515B
		HiStmFlo SI	14/15	II	7	513
		MismatchTrip	7	II	7	511B
	516	ΔPSI $P_1<P_4$	14/15	IV	16	516C
		$P_4<P_1$	14/15	IV	16	516D
		LoStm P SI	14/15	IV	16	516A

APPENDIX 1 (Cont'd)

INSTRUMENT CHANNEL DESCRIPTION		CHANNEL NO. DESCRIP.	TECH SPECS ACTION NO.	PROTEC. CHANNEL	RACK NO.	BISTABLE SWITCH NO. BS -
Steam Line 2 Pressure	524	ΔPSI $P_1<P_2$	14/15	I	3	514A
		ΔPSI $P_2<P_1$	14/15	I	3	514B
		HiStmFlo SI	14/15	I	3	522
		MismatchTrip	7	I	3	520B
	525	ΔPSI $P_2<P_4$	14/15	II	9	525A
		ΔPSI $P_4<P_2$	14/15	II	9	525B
		HiStmFlo SI	14/15	II	9	523
		MismatchTrip	7	II	9	521B
	526	ΔPSI $P_2<P_3$	14/15	III	11	526C
		$P_3<P_2$	14/15	III	11	526D
		LoStm P SI	14/15	III	11	526A
Steam Line 3 Pressure	534	ΔPSI $P_4<P_3$	14/15	I	5	534B
		ΔPSI $P_3<P_4$	14/15	I	5	534A
		HiStmFlo SI	14/15	I	5	532
		MismatchTrip	7	I	5	530B
	535	ΔPSI $P_2<P_3$	14/15	II	7	515A
		ΔPSI $P_3<P_2$	14/15	II	7	515B
		HiStmFlo SI	14/15	II	7	533
		MismatchTrip	7	II	7	531B
	536	ΔPSI $P_2<P_3$	14/15	III	11	526C
		ΔPSI $P_3<P_2$	14/15	III	11	526D
		LoStm P SI	14/15	III	11	536A
Steam Line 4 Pressure	544	ΔPSI $P_4<P_3$	14/15	I	5	534B
		ΔPSI $P_3<P_4$	14/15	I	5	534A
		HiStmFlo SI	14/15	I	5	542
		MismatchTrip	7	I	5	540B
	545	ΔPSI $P_2<P_4$	14/15	II	9	525A
		ΔPSI $P_4<P_2$	14/15	II	9	525B
		HiStmFlo SI	14/15	II	9	543
		MismatchTrip	7	II	9	541B
	546	ΔPSI $P_1<P_4$	14/15	IV	16	516C
		ΔPSI $P_4<P_1$	14/15	IV	16	516D
		LoStm P SI	14/15	IV	16	546A

INSTRUMENT CHANNEL DESCRIPTION		CHANNEL NO. DESCRIP.	TECH SPECS ACTION NO.	PROTEC. CHANNEL	RACK NO.	BISTABLE SWITCH NO. BS -
First Stage Pressure	505	HiStmFlo SI	14/15	I	3	512
				I	3	522
				I	5	532
				I	5	542
		P-7	8b	I	4	505A
First Stage Pressure	506	HiStmFlo SI	14/15	II	7	513
				II	9	523
				II	7	533
				II	9	543
		P-7	8b	II	8	506A

Inoperability of Turbine trip-reactor trip, SI-Reactor Trip, Logic, RCP undervoltage, underfrequency, seismic trip and breaker position devices would not normally be ascertained by Operations personnel, therefore the tripping procedure for an inoperable device is not included. The technician who discovered the problem would place the device in the tripped condition upon orders from the Shift Foreman if required by the TS Action statement.

PG&E **Pacific Gas and Electric Company**

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DATE	2/16/82
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DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

TITLE: EMERGENCY OPERATING PROCEDURE
ANTICIPATED TRANSIENT WITHOUT TRIP (ATWT)

APPROVED: R C Thornberry 2/13/82
PLANT MANAGER DATE

SCOPE

This procedure describes the steps to be taken in the event of an ATWT. An ATWT is a failure of the reactor protection system to trip the rods in when one or more reactor trip setpoints have been reached.

SYMPTOMS

1. Reactor trip point exceeded without a reactor trip.
2. Possible Reactor Protection System activiated alarm.
3. Possibly the reactor trip alarm.
4. DRPI indicates no rods drop.
5. RCS Hi puressure and level alarm.
6. NIS continues to read upscale.

AUTOMATIC ACTIONS

1. PZR PORVs open.
2. PZR spray valves open.
3. PZR safety valves open.
4. Steam dump activated.

OBJECTIVES

1. Ensure the reactor is shutdown.
2. Provide a heat sink for the reactor.

IMMEIDATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Manually trip the reactor.	1. Use the red handle.
a. Verify rod bottom lights on DPRI.	
b. Verify NIS decreasing.	

ACTIONS	COMMENTS
2. If the rods fail to drop after Step 1 above, open the 480 Volt LC 13 D and E breakers 52 HD 13 and 52 HE-4.	2. This will deenergize the load centers supplying power to the rod control MG sets.
3. If the rods fail to drop after Step 2 above, close the BIT Recirc. Valves (8807A and B, and 8911), open the BIT inlet and outlet valves (8803A and B, and 8801A and B) and start both centrifugal charging pumps.	3. If the BIT is injected and the rods remain out of the core, it is important to keep the RCP's in service and maintain hot standby conditions. A cooldown could allow the reactor to return to criticality.
4. Trip the turbine manually if required.	4. With the reactor protection system failed, the P-4 signal is not present to trip the turbine.
5. If the turbine fails to trip after Step 4 above, trip the turbine using the trip lever on the turbine pedestal.	
6. Verify all three auxiliary feedwater pumps running.	6. Manually start pumps if required.

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. Sound the Site Emergency Alarm.	
2. Verify steam dump operating to the condenser or 10% atmosphere steam dumps open. Transfer steam dump to the steam pressure mode with a 1005 psig setpoint.	2. Monitor the heat sink (steam dump) closely after this transient.
3. Verify that at least one RCP is operating. If not, start as many as possible.	3. RCP seals should be observed closely as the RCS Hi pressure may have affected them.
4. Check all rod bottom lights on, emergency borate 100 ppm for any stuck out rod. If no rods have inserted, emergency borate the RCS until 2000 ppm is achieved.	4. RCS Boron concentration is increased approximately 100 ppm with 45 gpm boration flow for approximately 6 to 7 minutes.

ACTIONS	COMMENTS
5. Check the gross failed fuel detector for any signs of fuel damage.	
6. Monitor steam generator water levels, air ejector off gas and steam generator blowdown radiation monitors for any indication of a steam generator tube rupture.	6. The leak may occur as a result of the RCS Hi pressure during the transient.
7. Monitor RCS parameters.	
a. Tavg should return to 547°F.	
b. PZR level should remain above 22%.	
c. RCS pressure should remain above 1950 psig.	
8. If pressurizer pressure decays below 1900 psig:	
a. Verify closed all PORV's (close the backup valve if a PORV is found open.)	a. The Hi pressure transient may have stuck open a PORV.
b. Verify closed the PZR spray valves. (Close any valve found open.) If the valve will not close, trip the associated RCP to prevent further spray.	
9. Monitor all SI initiation parameters (PZR pressure, containment pressure, etc.) for SI conditions. If any parameter exceeds the SI initiation setpoint, manually initiate safety injection and proceed to OP-0.	9. With a failure in the Reactor Protection System, the automatic SI initiation is in doubt.
10. If manual initiation of SI fails, proceed to OP-0 and perform all Immediate Operator Actions Steps using manual control.	

ACTIONS	COMMENTS
11. If SI is not required, proceed as follows:	
a. Verify closed or close feedwater control valves when Tavg reaches 554°F.	a. If the P-4 signal failed, auto closure of these valves may not occur.
b. Transfer the NIS recorder to monitor one IR and one SR channel.	
c. Check the turbine-generator coasting down properly.	
1) All turbine drain valves open.	
2) The AC bearing oil pump and the high pressure seal oil backup pump start automatically.	
3) The lift pump starts at about 600 RPM.	
4) The turning gear engages automatically at or near zero speed.	
d. Maintain condenser vacuum; if vacuum is lost, use the 10% atmospheric dump valves to control steam generator pressure.	
e. Establish and maintain hot standby operation, verify shutdown margin per STP R-19 (Shutdown Margin Calculation), and adjust RCS Boron concentration if necessary.	
f. If condenser vacuum is lost, check the level in the condensate storage tank to determine how long the unit can be maintained in hot standby prior to going to cold shutdown.	
g. Prepare to take the plant to cold shutdown conditions.	

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   If plant conditions indicate the required coincidence for reactor trip has occured, or the required coincidence of bistables have tripped or trip is manually activated, designate the event according to subsequent plant conditions as follows:

   Alert - NIS indicates reactor not subcritical (non-negative startup rate following trip).

   Site Area Emergency - Power range channels indicate continued power operation.

   General Emergency - Power range channels indicate continued power operation and fuel damage as a consequence of this event is evident (sample indicates > 300 μCi/CC equivalent of I-131 or > 1% fuel failures in 30 minutes or 5% total fuel failures).

   Notify plant staff and response organizations required for this classification by implementing Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

TITLE: EMERGENCY OPERATING PROCEDURE
ACCIDENTAL DEPRESSURIZATION OF MAIN STEAM SYSTEM

APPROVED: R C Thornberry 2/19/82
PLANT MANAGER DATE

SCOPE

This procedure covers the operational steps to be taken in the event of one or more safety or steam dump valves spuriously opening while at Hot Standby or at power conditions.

SYMPTOMS

1. Sudden increase in steam flow.
2. Sudden rise in one or all steam generators levels.
3. Decrease in Tavg.
4. Decrease in Pressurizer Pressure and level.
5. Possible reactor trip on PZR Lo Pressure.
6. Possible Safety Injection on PZR Lo Pressure.

AUTOMATIC ACTIONS

1. Possible reactor trip on PZR Lo Pressure.
2. Possible safety injection on low pressurizer pressure or steam line hi ΔP.

OBJECTIVES

1. To minimize the transient.
2. To prevent a reactor trip or SI if possible.

IMMEDIATE OPERATOR ACTIONS

ACTIONS	COMMENTS
1. If the reactor trips, go to Emergency Procedure OP-5.	
2. If an SI initiated, go to Emergency Procedure OP-0.	

SUBSEQUENT OPERATOR ACTIONS

ACTIONS	COMMENTS
1. If the steam dump valve indicator indicates an open steam dump valve, go to OFF RESET position on the steam dump control switches.	1. This should close any steam dump valve with a control malfunction.
2. If a 10% steam dump is open and Step 1 above fails to close it, increase the pressure setting on the individual controller.	
3. If Steps 1 or 2 above fail to close the valve, proceed to the valve and close the manual stop upstream of the open valve.	
4. If a safety valve has lifted and fails to reseat, trip the reactor and verify or initiate an SI signal. Proceed to Emergency Procedure OP-0 (Reactor Trip with Safety Injection).	4. The results of an open safety valve are the same as a steam line break.

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. Designate this event a Notification of Unusual Event. Notify plant staff and response organizations required for this classification by implementing Emergency Procedures G-2 "Establishment of the On-Site Emergency Organization and G-3 "Notification of Off-Site Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP OP-41
REVISION 1
DATE 2/16/82

TITLE: EMERGENCY OPERATING PROCEDURE
HYDROGEN "EXPLOSION" INSIDE CONTAINMENT

APPROVED: [signature] 2/15/82
PLANT MANAGER DATE

SCOPE

This emergency procedure provides guidance in identifying and mitigating the consequences of a hydrogen gas burn or "explosion" inside containment. It is assumed that a Loss of Coolant accident has occurred.

SYMPTOMS

1. Short time duration containment pressure spike.
2. Reduction in containment H_2 concentration as measured by H_2 monitors on the Post-Accident Monitor Panel after the pressure spike.
3. No significant increase in containment radiation levels.
4. No significant change in RCS pressure or conditions.

AUTOMATIC ACTION

1. Possible actuation of Phase B isolation and containment spray.

OBJECTIVE

1. Determine containment integrity.
2. Prevent further hydrogen explosions.

IMMEDIATE OPERATOR ACTION

COMMENTS

1. Verify H_2 recombiners operating.
2. Continue with Emergency Procedure OP-0 or OP-1. Verify adequate core cooling to prevent further H_2 production using Appendix F in Emergency Procedure OP-1.
3. Monitor containment pressure recorder. If pressure returns to near original levels in a short time period after the peak,

and

Containment spray was initiated by this

IMMEDIATE OPERATOR ACTION	COMMENTS
pressure spike, reset phase B isolation and terminate containment spray if containment pressure is <22 psig.	
4. Reestablish CCW to the RCP's, via FCV-355, 365, 357, 363, 749 and 750 if Phase B isolation occurred.	4. If RCP's are still running, shutdown the RCP's within 5 minutes of the CCW isolation if CCW cannot be reestablished.
5. If a hydrogen "explosion" is indicated per the above, and containment pressure rapidly decays below the pre-explosion level, assume containment integrity lost, reinstate Phase B isolation and declare a general emergency.	

SUBSEQUENT OPERATOR ACTIONS

1. Continue to monitor containment pressure, containment H_2 concentration, and changes in radiation levels outside containment, which may be indicative of containment leakage.
2. To prevent further H_2 generation, a constant ongoing check must be made on Adequate Core Cooling. Continue to use Emergency Procedure OP-1 Appendix F (Determination of Adequate Core Cooling).

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Site Emergency Coordinator, proceed as follows:

   a. Designate this event a General Emergency. Notify plant staff and response organizations required by Emergency Procedures G-1 "Establishment of the On-Site Emergency Organization" and G-3 "Notification of Off-Site Organizations" and implement the instructions in Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation" regarding on and off-site protective actions.

PG&E **Pacific Gas and Electric Company**

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

NUMBER EP R-1
REVISION 6
DATE 3/2/82

TITLE: EMERGENCY PROCEDURE
PERSONNEL INJURY (RADIOLOGICALLY RELATED) AND/OR OVEREXPOSURE

APPROVED: R C Thornberry 3/3/82
PLANT MANAGER DATE

SCOPE

This procedure describes the actions which are to be taken in the event of:

1. Personnel injury (minor or serious) where the victim is radiologically contaminated.
2. Overexposure (or suspected overexposure) from an external source.
3. Overexposure (or suspected overexposure) from an internal source.
4. A combination of the above.

Injuries which do not involve radioactive contamination or overexposure are handled in accordance with Emergency Procedures M-1 or M-2.

DISCUSSION

Any radiologically related injury or potential radiation overexposure is a serious matter requiring prompt attention to the care of the injured and prompt appropriate corrective action to preclude re-occurrence. In addition, followup investigation to quantify the extent of exposure to radiation requires care in the gathering and retention of samples, radiation readings and other evidence which may contribute to the understanding of the incident and assist both in care of the injured and in preventing re-occurrence.

IMMEDIATE ACTIONS

1. The employee(s) who are at the scene shall:
   a. Render all necessary first aid.
   b. Notify the control room (Shift Foreman) as soon as practical.
2. Shift Foreman (Interim Site Emergency Coordinator)
   a. Evaluate plant status that may have produced the personnel injury and/or overexposure. Sound the site emergency signal to clear the affected area, if the situation warrants it.

b. Dispatch additional personnel to the scene of the injury if required.

c. Call an ambulance if the injury warrants it. Refer to Appendix 1 "Measures to be taken if Medical Care Is Required" for instructions.

SUBSEQUENT ACTIONS

The Shift Foreman shall direct all subsequent actions until relieved by the long-term Site Emergency Coordinator if the situation warrants it.

1. Actions Common to All Occurrences

a. Transport the patient to the first aid room, provided that this can be done without aggravating the injury.

b. Take actions as specified in the following sections as appropriate for the particular occurrence.

Section 2: Minor injury when contamination is present.

Section 3: Serious injury when contamination is present.

Section 4: Overexposure from external source

Section 5: Overexposure from internal source.

c. Perform the notifications required by Appendix Z "Emergency Procedure Notification Instructions."

NOTE: Form 69-9221 "Emergency Notification Record" is provided to record notifications not documented elsewhere.

d. Begin gathering information to assist the long-term Site Emergency Radiological Advisor in his evaluation. Guidance on things which should be investigated is given in Appendix 2 "Factors to Consider in Making a Preliminary Evaluation."

e. Close out the event with the following written reports:

1) Report to NRC (required within 24 hours for an Unusual Event, or within 30 days for a report under 10CFR20.403).

2) Form 62-4587 "Report of Industrial Injury to Employee."

3) Form 62-4586 "Employers' Report of Occupational Injury or Illness."

4) Nuclear Plant Problem Report. (See Nuclear Plant Administrative Procedure C-12.)

NOTE: Reports to NRC and the Nuclear Plant Problem Report are not required for minor injuries for which onsite first aid and decontamination is adequate.

2. Minor Injury When Contamination is Present

The following steps apply to injuries where prompt medical attention is not required (i.e., first aid at the plant is adequate).

a. Make the following surveys and record the results on the "Skin and Clothing Decontamination" Form (Form 69-9392).

1) The wound prior to decontamination.

2) The object causing the injury (if possible) and any clothing penetrating or touching the injury. These items should be retained, if possible, until the long-term Site Emergency Radiological Advisor has completed his evaluation so that detailed radionuclide analysis can be performed, if required.

b. Decontaminate the wound using the standard procedures discussed in Radiation Control Procedure G-4. In cases of severe contamination, where there is a realistic possibility that significant ingestion of radionuclides may have occurred, it is desirable to retain wash solutions (or samples thereof), swabs, and other such material which may be useful to the Site Emergency Radiological Advisor.

NOTE: Refer to Emergency Procedure RB-5 "Personnel Decontamination" in the event normal decontamination facilities are overloaded or unavailable.

c. When the wound is clean, resurvey and record the results on a survey form.

d. Complete any additional first aid measures.

e. Complete accident report Form 62-4587, "Report of Industrial Injury to Employee" and forward to plant clerk for processing.

NOTE: This documentation requirement assumes no medical attention (beyond first aid) is required and that no lost time occurs. If lost time beyond the day of injury is likely, or if medical treatment (including doctor referral) is required, complete Form 62-4586, "Employers' Report of Occupational Injury of Illness" and forward to plant clerk.

3. Serious Injury When Contamination is Present

The following steps apply to injuries where prompt medical attention is required (i.e., the patient must be taken to a hospital) and the patient is contaminated. In this type of circumstance, the need for treatment of the injury and comfort of the patient will take precedence over the need for decontamination.

a. Call San Luis Ambulance and French Hospital and have the patient transported to French Hospital. The detailed steps to be taken if this is required are given in Appendix 1 of this procedure.

b. During the interval until the ambulance arrives keep the patient as comfortable as possible. Survey and decontaminate the patient to the extent that time and conditions permit. Do not decontaminate the patient if it will aggravate his injury. Record survey results on the "Skin and Clothing Decontamination" Form (Form 69-9392.)

1) Survey any wounds and/or the victim's skin (if possible).

2) Survey the object causing the injury (if possible) and any clothing penetrating or touching the injury. These items should be retained, if possible, until the long-term Site Emergency radiological Advisor has completed his evaluation so that detailed radionuclide analyses can be performed, if required.

3) Decontaminate the patient using the standard procedures discussed in Radiation Control Procedure G-4. In cases of severe contamination, where there is a realistic possibility that significant ingestion of radionuclides may have occurred, it is desirable to retain wash solutions (or samples thereof), swabs, and other such material which may be useful to the Site Emergency Radiological Advisor.

NOTE: Refer to Emergency Procedure RB-5 "Personnel Decontamination" in the event normal decontamination facilities are overloaded or unavailable.

c. Have the hospital kit and a handheld radio available for transport to the hospital with the monitor accompanying the patient, or the team dispatched to the hospital.

4. Overexposure From External Source

The following steps apply to cases where the patient has (or is suspected to have) received a dose from an external source to the whole body, or any portion thereof, in excess of an applicable limit contained in Radiation Control Standard No. 1, and where the individual does not require prompt medical attention for any other reason. Personnel suspected of overexposure shall not re-enter radiation controlled areas unless authorized by the Site Emergency Coordinator.

a. Provide any first aid or medical attention which the patient may required.

b. Notify San Luis Ambulance and French Hospital and transport the patient to French Hospital in accordance with Appendix 1 for observation or treatment in any of the following circumstances:

1) The patient is known or suspected to have received at least any of the following:

   a) 25 rem to the whole body, active blood forming organs, lens of eyes, gonads, head or trunk.

   b) 150 rem to the skin.

   c) 375 rem to the extremities.

2) The patient shows signs of radiation sickness, such as nausea, vomiting, extreme sweating, weakness, diarrhea, extreme anxiety, incoherence, sensitivity of the nerves (tingling or itching sensation).

3) The patient shows evidence of radiation dermatitis (skin damage). Except for extremely high skin dose (greater than 5,000 rem), in which case pain occurs promptly and is intense, the symptoms at the time of exposure are a sensation of warmth and itching. Redness, blistering and other effects may not appear for several days.

c. If the patient requires transportation to the hospital, during the interval until the ambulance arrives keep the patient comfortable. Survey the individual and perform any decontamination which circumstances require and/or permit. <u>Do not aggravate any injury or unduly alarm the patient in performing these operations</u>. Record survey results on the "Skin and Clothing Decontamination" Form (Form 69-9392) and/or "Radiation Dose Rate Survey Record" (Form 9316). In cases of severe contamination, handle as in Step 3.c to the extent practical.

d. To the extent practical, save all vomit, urine, feces or other samples which may assist the long-term Site Emergency Radiological Advisor in evaluating the accident. This is particularly important if significant ingestion of radioactive materials is suspected.

e. Collect the patient's personnel dosimetry prior to sending him to the hospital or releasing him. This will be processed for evaluation.

f. Subsequent actions will be based upon the results of the evaluation of the external exposure.

5. Overexposure From Internal Sources

The following steps apply to cases where the patient has (or is suspected to have) ingested a significant quantity of radioactive material. If the ingestion was by breathing, this procedure applies any time that the concentration to which the person has been exposed is greater than or equal to (MPC) x PF, where (MPC) refers to the normal (40 hr.) maximum permissible concentration, and PF refers to the protection factor of any respirator that the patient was wearing.

a. Take any medical action which may be required as a result of injury or external dose received (Steps 3 and 4 above). The treatment of these effects should take precedence over the evaluation of internal exposure.

b. Remove and retain for subsequent radiological analysis the patient's clothing and respirator.

c. Survey the patient thoroughly and record the results on the "Skin and Clothing Decontamination" Form (Form 69-9392).

d. Thoroughly decontaminate the individual. If practical, save samples of the decontamination solutions, swabs, and other materials which may be of use in subsequent radiological evaluations.

e. Count the patient on the whole body counter. The results of this analysis will, in large measure, determine the necessity for further medical attention or surveillance.

f. Collect and save any urine, feces, or vomit which is passed from the patient. The long-term Site Emergency Radiological Advisor may request that special urine samples be collected for bioassay.

g. Subsequent actions will be based upon the results of the evaluation of the internal exposure.

h. If the patient is sent to the hospital, make arrangements to have all urine, feces or vomit samples retained for radiological analysis.

REFERENCES

1. Radiation Control Standard No. 1, "Personnel Exposure."
2. Radiation Control Standard No. 2, "Internal Exposure Controls."
3. Radiation Control Standard No. 5, "Medical."
4. Radiation Control Standard No. 8, "Reporting Requirements."
5. Radiation Control Procedure No. G-3, "Personnel Internal Exposure Control."
6. Radiation Control Procedure No. G-4, "Personnel Contamination Control."
7. Radiation Control Procedure No. G-7, "Radiation Surveys."
8. Emergency Procedure G-1, "Accident Classification and Emergency Plan Activation."
9. Emergency Procedure G-2, "Establishment of the Onsite Emergency Organization."
10. Emergency Procedure G-3, "Notification of Offsite Organizations."
11. Emergency Procedure R-4, "High Radiation (In Plant).
12. Emergency Procedure RB-5, "Personnel Decontamination."

ATTACHMENTS

1. Form 69-9221, "Emergency Notification Record."
2. Form 69-9316, "Radiation Dose Rate Survey Record."
3. Form 69-9392, "Skin and Clothing Decontamination."
4. Form 62-4587, "Report of Industrial Injury to Employee."
5. Form 62-4586, "Employers' Report of Occupational Injury or Illness."
6. Form 62-6015, "Medical Referral."
7. Light Duty Program Letter.
8. Safety, Health and Claims Personnel to Be Contacted for Reporting of Injuries at Diablo Canyon (6/81).

APPENDIX 1

MEASURES TO BE TAKEN IF MEDICAL CARE IS REQUIRED

The following are the procedural steps to be taken in the event a contaminated patient must be transported to the hospital for medical treatment:

1. Call San Luis Ambulance (Phone 543-2626) and provide the following information:

   a. Name of caller.

   b. Company affiliation.

   c. Phone number of caller. (Where he can be reached.)

   d. Name of injured person.

   e. Where he is located.

   f. Where he is to be transported (French Hospital).

   g. Nature of injury.

   h. Patient is contaminated.

   i. Any other medical information which might be pertinent to transporting the injured person.

   Record this information on Form 69-9221, "Emergency Notification Record."

2. Contact the security force at the Port San Luis entrance and alert them that the ambulance is entering. It is also advisable to have an escort accompany the ambulance to the first aid room to minimize the delay in reaching the destination.

3. The victim shall be transported to French Hospital. Call ahead to the hospital (Phone 543-5353) and provide the following information:

   a. Name of caller.

   b. Company affiliation.

   c. Phone number of caller. (Where he can be reached.)

APPENDIX 1 (Cont'd)

MEASURES TO BE TAKEN IF MEDICAL CARE IS REQUIRED

d. Name of injured person.

e. Age of injured person (approximate if not known).

f. Extent of injury or symptoms.

g. Medical history (if known).

h. Radiological conditions.

Record this information on Form 69-9221, "Emergency Notification Record."

4. Prior to arrival of the ambulance, the patient should be decontaminated to the extent practical without aggravation of injury.

5. If the patient cannot be completely decontaminated prior to arrival of the ambulance, wrap him in a blanket prior to placing him in the ambulance in order to minimize the spread of contamination. Alternatively, he may be placed in the plant's Nuclear Accident Emergency Carrier.

6. An individual qualified in radiation monitoring shall accompany the victim to the hospital. This individual should take a hospital kit and a handheld radio with him.

NOTE: Two hospital kits and radios are stored in the Security Building Weapons Storage Room. Request access from the Security Shift Supervisor.

7. Two additional individuals qualified in radiation monitoring should be dispatched to French Hospital to assist hospital personnel.

APPENDIX 2

FACTORS TO CONSIDER IN MAKING A PRELIMINARY INVESTIGATION

It is important to conduct the preliminary investigation in a systematic manner to assure that potentially valuable evidence is not overlooked, lost or destroyed. The following is a reference listing of items which should be checked (if they are applicable). Also, two other factors are important in conducting an investigation of this type, namely: a) information which is gathered should be written down in a comprehensive, neat manner, and b) all samples, clothing, or other articles which are collected should be put in sample bottles or plastic bags, and labelled with the patient's name, date, collection time, sample identification, and other pertinent data.

1. Factors Common to All Accidents

   a. Date, time of occurrence.

   b. Basic reconstruction of events.

   c. Probable source(s) of radioactivity involved.

   d. Names and addresses of all witnesses.

2. Considerations in Evaluating External Exposure

   a. Exactly where was the patient located at the time of exposure?

   b. How was patient physically oriented with respect to source (will help to evaluate nonuniform exposure)?

   c. On what part(s) of body were dosimeters being worn?

   d. Were self-reading dosimeter readings recorded and all nonself-reading types collected?

   e. Are there any "natural" dosimeters available? (Belt buckles, wrist watches, gold tooth fillings, and other such items are useful in determining neutron dose.)

   f. Exactly what was the time interval over which exposure occurred?

g. Are there any applicable dose rate measurements, and if so, exactly where and when were they made?

1) Ion chamber measurements

2) Area monitors

3) Other

3. Considerations in Evaluating Internal Exposure

a. Where was the patient located at time of exposure?

b. Exactly what was the time interval over which exposure occurred?

c. Can sample(s) of liquids which were ingested be obtained?

d. Can samples of airborne activity which were breathed be obtained before the area is purged?

e. Are there any applicable monitor readings?

1) Process monitors

2) Continuous Air Monitors

3) Area Monitors

) Other

f. Can samples of patient's clothing, decontamination solutions, secretions, respirator filters, be saved?

g. Can the region in the vicinity of the occurrence be smear-tested, or can decontamination solutions be retained?

APPENDIX Z

EMERGENCY PROCEDURE NOTIFICATION INSTRUCTIONS

1. When this emergency procedure has been activated and upon direction from the Shift Foreman, proceed as follows:

   a. In case of a minor injury with contamination present or an overexposure case from any source which does not meet the criteria for an Unusual Event, notify the Plant Manager, Plant Superintendent and Supervisor of CHemistry and Radiation Protection or their designated alternates.

   b. Designate this event a significant event in a case of overexposure from an external source where the exposure (for the quarter) exceeds the following:

      5 Rem Whole Body
      30 Rem Skin of Whole Body
      75 Rem Extremities

      Notify the NRC Bethesda Operations Center using the red phone in the Control Room as a minimum within one hour. Gather sufficient information from all sources prior to calling so that the phone call is meaningful. Refer to Operating Procedure O-4 "Operating Order (One Hour Reporting Requirements to NRC)" for a suggested format for reporting. Notify the NRC that your call is pursuant to 10CFR50.72 (Notification of Significant Events).

      Notify the Director, NRC Region 5, by telephone and telegraph, mailgram or facsimile within 24 hours of the event. Indicate the notification is pursuant to 10CFR20.403 (Notification of Incidents).

   c. Designate this a Notification of Unusual Event in any case of an injury or overexposure requiring transportation of the patient to an offsite hospital or if extensive onsite decontamination is required (soap and water washings do not remove contamination or offsite decontamination assistance is required). Notify plant staff and response organizations required for this classification by implementing Emergency Procedures G-2 "Establishment of the Onsite Emergency Organization" and G-3 "Notification of Offsite Organizations" in accordance with Emergency Procedure G-1 "Accident Classification and Emergency Plan Activation."

d. In addition to the notifications performed under "c." above, for a Notification of Unusual Event, if the case involves an overexposure from an external source which exceeds:

25 Rem Whole Body
150 Rem Skin
373 Rem Extremities

Immediately notify the Director, NRC Region 5 by telephone and telegraph, mailgram and facsimile. Indicate the notification is pursuant to 10 CFR20.403 (Notification of Incidents).

2. In addition to notification performed above, also notify the following in any case where NRC notification is required.

a. Supervising Nuclear Generation Engineer (Personnel and Environmental Safety) or his alternate in the Department of Nuclear Plant Operations:

Mr. S. M. Skidmore PGandE:
Plant Ext.:
Home:

b. Compensation Claims Representative in the Department of Safety, Health and Claims, per the attached list of personnel.

NOTE: 1) The System Dispatcher will handle the notification of General Office Personnel if they cannot be promptly reached.

2) Nuclear Mutual Limited (NML) holds the Company liability and property damage insurance for Company personnel and property. They should be notified under the same circumstances as the NRC. Notification is made by the Company's Insurance Department. The Department of Nuclear Plant Operations should be requested to interface between the plant and the Insurance Department when required. American Nuclear Insurers/Mutual Atomic Energy Liability Underwriters (ANI/MAELU) holds third party insurance coverage and would be similarly notified in accidents involving a third party.

69-9221 7/80 (100)

DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT

EMERGENCY NOTIFICATION RECORD

EMERGENCY IDENTIFICATION ______________________ DATE __________ SHEET ________

PERSON CALLED	AFFILIATION	TIME	REACHED	BY	MESSAGE GIVEN	RESPONSE

69-9316 7/80 (100)

DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT

RADIATION DOSE RATE SURVEY RECORD

DATE ____________ TIME __________ SWP/RWP NO ______________ SURVEY NO. ________

AREA OR EQUIPMENT ____

TYPE OF SURVEY ______________________________________ SHEET ____ OF ____

ITEM NO.	DESCRIPTION	BETA		GAMMA		NEUTRON	
		m rad/hr	distance	mR/hr	distance	mrem/hr	distance

SURVEYED BY ____

SURVEY TYPE	INSTRUMENT	TYPE DETECTOR	SERIAL NO.	DATE CALIB. DUE
BETA				
GAMMA				
NEUTRON				

COMMENTS ____

RECOMMENDATIONS ____

SURVEY REVIEWED ______________________ DATE ________
SUPERVISOR

69-9392 7/81 (100)

DIABLO CANYON POWER PLANT

SKIN AND CLOTHING DECONTAMINATION

NAME	DATE	TIME OF OCCURRENCE AM PM

UNIT	ELEV.	BLDG.	ROOM	AIR SAMPLE LOG NUMBER	SURVEY LOG NUMBER

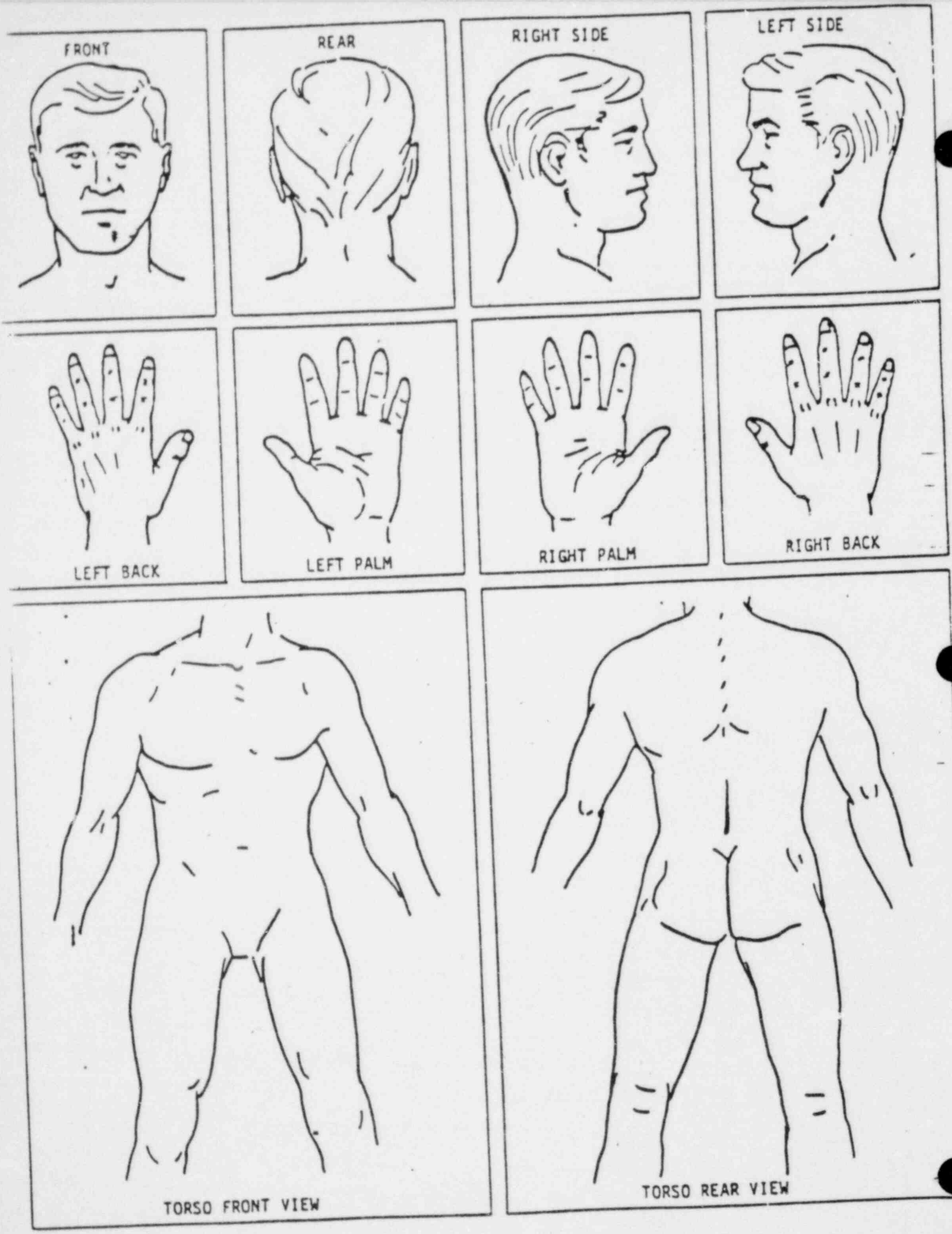
ITEM NO.	LOCATION OF CONTAMINATION	TIME OF SURVEY	INITIAL SURVEY RESULTS	SURVEY INSTRUMENT

DECONTAMINATION INFORMATION

ITEM NO.	DECONTAMINATION AGENT USED	TIME	SURVEY INSTRUMENT	SURVEY RESULTS	SKIN CONDITION AFTER TREATMENT

REMARKS

SIGNATURE:



☐ SKIN ☐ PROTECTIVE CLOTHING ☐ PERSONAL CLOTHING

62-4587 Rev. 1/79

PACIFIC GAS AND ELECTRIC COMPANY

Report of Industrial Injury to Employee

1. Name ______________________ 6. Division ______________________

2. Address ______________________ ZIP __________

3. Telephone No. ______________________ 7. Department ______________________

4. Social Security No. ______________________ 8. Date of Accident ______________________

5. Occupation ______________________ 9. Time of Accident ______________________

10. Location of Accident ______________________ 11. Nature of Injury ______________________

12. What were you doing and how did accident occur? ______________________

____

____

13. Describe First Aid rendered: ______________________

14. Witnesses to accident:

1. ______________________

2. ______________________

3. ______________________ 15. ______________________
Signature of Employee

16. Date injury reported: ______________________

17. Date 30 days elapses: ______________________ 18. ______________________
* See Over
Signature of Supervisor

INSTRUCTIONS: This report (items 1 thru 15) should be *written* and *signed* by the *employee personally* and countersigned by the supervisor. It is for all Industrial Injuries and is in duplicate. The original is to be retained for Company records; the copy is to be detached after completion and given to the employee. Before signing in Item 18, the *supervisor* should fill in the date of the report (Item 16) and compute and notate the date *30 days* from the date the injury was reported (Item 17).

If the employee later requires treatment by a doctor or becomes disabled, Form 62-4586 must be prepared and forwarded to the Safety, Health and Claims Department *IMMEDIATELY* accompanied by the original of this report.

If the employee is unable to fill out or sign this report, it should be prepared, signed by the supervisor and the employee should be given a copy within 5 days as required by law.

If the injured employee cannot write English, the report may be made according to a verbal statement. If necessary, the employee may sign by a mark and a witness to the report should sign below the employee's mark.

INFORMATION FOR THE INJURED EMPLOYEE

This notice complies with the California Labor Code

I. **General Information:** The Company has an extensive safety program to help its employees avoid injury. In the event of a work-related injury requiring medical care, special provision has been made for the best medical services available. The Company is very much concerned with its injured employees and is proud to extend the medical program developed over years of experience for your benefit. Every reasonable effort will be devoted in minimizing the extent and duration of your industrial injury.

The Company is entirely self-insured for industrial injuries to its employees which arise out of and occur in the course of employment. All compensation benefits, including medical treatment, rehabilitation programs, and disability payments are administered by the Company. If questions arise, please contact your supervisor.

II. **Medical Benefits:** Through continuing efforts, the Company has utilized the talents of highly qualified physicians and specialists throughout PG&E system. A panel of doctors familiar with the various Company programs and benefits, including the light duty work program, has been established to provide a greater service to the injured employee.

You are entitled to receive medical, surgical, and hospital services and supplies reasonably required to cure or relieve you from the effects of your injury, including nursing care and such things as crutches and artifical limbs. Reasonable transportation expense incidental to treatment will also be provided.

III. **Selection of Treating Physician:** Treatment of industrial-injured employees is provided by the employer at the employer's expense with the employee having the opportunity to change physicians if desired. California law permits employees who sustained an industrial injury to be treated by a physician or at a facility of their choice within a reasonable geographic area *commencing 30 days after the date injury is reported*, or immediately by your personal physician, provided you notified the Company prior to your injury.

If you wish to continue your present treatment, you may do so. It is recommended that you continue with the physician that has been provided, but if you wish to change doctors, notify your supervisor. The Company's experience in this area is available to assist you in selecting the proper medical care. If you elect to change to another treating physician or facility after 30 days, you must notify your supervisor of the name and address of the physician or facility you have selected to continue treatment. You should show this document to the physician or facility so they will be notified of the immediate duty to report to the Company as required by *Section 4603.2* of the Labor Code. If the facility or physician requests, you are required to sign a medical information release to permit reports of treatment to be rendered to the Company.

IV. **Amount of Indemnity Payable:** If your weekly wage exceeds $231.00, you are entitled to the maximum Temporary Disability indemnity of $154.00 per week, commencing on the 4th full day after injury. If the work-related injury results in hospitalization or more than 21 days of disability, payments will commence the 1st full day of disability. If your disability results in lost time for over two years or you lose time after two years, you will be paid temporary disability at the rate currently in effect. This applies only to injuries on or after 1-1-75. Permanent disability is paid at the rate of $70.00 per week.

V. **Rehabilitation:** Effective January 1, 1975, the employer must provide a rehabilitation program for any employee where the treating physician advises the Company that the employee will be unable to return to his usual and customary occupation at the time of injury, on a permanent basis.

This program provides services such as vocational evaluation, counseling, retraining, including on-the-job training and placement necessary to restore the injured employee to suitable employment, which is not confined to reemployment with PG&E. The Company works in conjunction with the California Rehabilitation Bureau.

VI. **Death Benefits:** If your injury results in death and you have a totally dependent spouse, the sum of $50,000.00 is the maximum benefit, except in cases involving a spouse and one or more dependent minor children, the maximum is $55,000.00. There is also a maximum burial allowance of $1,500.00. In cases of partial dependency, the death benefit will be a sum equal to four times the amount annually devoted to the support of the dependents not to exceed $50,000.00.

VII. **Further Information:** If you wish further information on your particular case, in addition to what your supervisor has provided, contact the Workers' Compensation Claims Section (415) 781-4211 Extension 3171.

Information and Assistance Officers located in the offices of the Division of Industrial Accidents, Workers' Compensation Appeals Board are a further source of information and services. The Workers' Compensation Appeals Board is the final arbiter of claims to workers' compensation.

62-1544 Rev. 3/79

PACIFIC GAS and ELECTRIC COMPANY
Employer's Report of Occupational Injury or Illness
CONFIDENTIAL — *For Use by Company Attorneys*

DIVISION
GENERAL OFFICE OR
GENERAL CONSTRUCTION

DEPARTMENT | DISTRICT, TOWN OR LOCAL OFFICE | R. C. NUMBER

LOCATION OR ITEM NUMBER | ACCOUNT NUMBER | JOB NUMBER

ACCIDENT REPORT NUMBER

ALPHA	YEAR	NUMBER	O.S.H.A. ESTB. CODE
1	2 3	4 8	

California law requires an employer to report within five days every industrial injury or occupational disease which (a) results in lost time beyond the day of injury, or (b) requires medical treatment other than first aid. These must be reported to the Safety, Health and Claims Department, General Office within three days so that the Company can comply with the law. In addition, cases which result in death or (a) require hospitalization of more than 24-hrs. for other than observation, (b) result in loss of any member of the body, or (c) produce any serious degree of permanent disfigurement, require an immediate telephone report to the Department so that the appropriate government agencies can be notified as required by law.

EMPLOYER

1. Firm Name: PACIFIC GAS AND ELECTRIC COMPANY
2. Mailing Address: 77 BEALE STREET, SAN FRANCISCO, CA 94106 Phone 781-4211, Ext. 3171
3. Address and Phone Number of Reporting O.S.H.A. Establishment
4. Nature of Business: PUBLIC UTILITY — Gas & Electric
5. Unemployment Insurance Account Number: 002-2199

EMPLOYEE

6. Name
7. Social Security Number
8. Home Address
8a. Home Phone Number
9. Sex ☐ MALE ☐ FEMALE
10. Occupation / Job Title
11. Age
12. Department
13. Wages $______ ☐ PER WEEK ☐ PER MONTH
13a. Length of Service with P.G. & E. Co. ______ YEARS
13b. How long has employee been employed in his present occupation? A ☐ Less than Six Months B ☐ From Six Months to Two Years C ☐ Over Two Years
13c. Years of Experience

INJURY OR ILLNESS

14. Where did accident occur? Number and Street / City or Town / County
15. On Employer's Premises ☐ YES ☐ NO
16. What was employee doing when injured? (Be specific)
17. How did accident or illness occur? (Use separate sheet if necessary)
18. Tool, object or substance that directly injured employee
19. Nature of injury or illness and part of body affected
20. Name and address of physician
21. Name and address of hospital, if hospitalized
22. Date of injury or illness Mo./Day/Yr.
23. Time of Day HOURS
24. Was employee unable to work on any day after incident? ☐ YES — Date last worked: ☐ NO
If answer to #24 is "Yes" has supplemental benefit form been signed? ☐ YES
25. Has employee returned to work? ☐ YES — Date returned: ☐ NO Still Off Work.
26. Did employee die? ☐ YES — Date: ☐ NO

Under whose direction does employee work?

Date accident first reported | Date of this report

Report completed by (print or type name and title)

Dept. 9	PLEASE DO NOT USE THIS COLUMN
Town 10	CASE No
Social Security Number 11-19	EMPLOYER No
Age 34-35	INDUSTRY
Date 36-41	SEX
Time 42-45	AGE
Occupation 46-48	OCCUPATION
Service 49	WEEKLY WAGE
Co. Service 50-51	COUNTY
52-54	ACCIDENT TYPE
55-57	AGENCY
58-60	AGENCY PART
61-63	SUPPLEMENTAL AGENCY
64-65	NATURE OF INJURY
66-67	PART OF BODY
Nat. Inj. 68-69	INJURY DATE
Pt. Body 70-71	EXTENT OF INJURY
Type Inj. 72	INSURANCE CARRIER
73	REPORT LAG
74	CODED BY
N.I. 75-76	
Card Code J 80	

Filing of this report is not an admission of liability. "... No report of injury required to be filed by an employer or insurer by this chapter shall be admissable as evidence in any adversary proceeding before the Workmen's Compensation Appeals Board."

Labor Code, Section 6412

FIRM: PACIFIC GAS & ELECTRIC COMPANY
SIGNED BY: Raymond W White
Raymond W. White
OFFICIAL POSITION: Mgr., Safety, Health & Claims Dept.
TELEPHONE: 781-4211 EXTENSION: 3171

If you wish to exercise your rights under item (1) of the information section, please separate this page and present it to your selected physician.

§ 9785. Duties of the Employee-Selected Physician. The physician or facility chosen by the employee who undertakes to provide treatment pursuant to Labor Code Section 4600 shall:

(a) Within 3 working days after undertaking to provide such treatment notify the employer of the name and address of such treating physician or facility, and

(b) Within 5 working days following initial examination shall submit a written report to the employer to include:

(1) The name and address of injured employee;

(2) The employee's medical history as obtained by the physician;

(3) Findings on examination;

(4) The subjective complaints reported by the employee;

(5) The planned course, scope and duration of treatment;

(6) If appropriate, the estimated return-to-work date;

(7) An opinion as to whether residual permanent disability is to be anticipated and, if possible, an estimate of its extent;

(8) An opinion as to whether the employee will eventually be able to engage in the occupation being performed at the time of injury.

(c) At reasonable intervals during active treatment submit progress reports to the employer and, particularly, report promptly to the employer when:

(1) The employee's condition permits return to work;

(2) The employee's condition requires him or her to leave work;

(3) Hospitalization or surgery is indicated or recommended;

(4) The employee's condition becomes permanent and stationary;

(5) The employee's condition undergoes a previously unexpected significant change; (this report shall contain a statement of the proposed course of treatment required, if any, by that change);

(6) The employee is referred to another physician for consultation;

(7) The employee reasonably requests additional appropriate information.

Report # __________ Date __________, 19__

Dr. __________

Kindly give bearer,

Mr./Ms. __________
medical attention, and forward a complete detailed report immediately to Manager, Safety, Health and Claims Dept., 245 Market Street, San Francisco. Your bills should be itemized and all bills and reports rendered in triplicate.

PACIFIC GAS AND ELECTRIC COMPANY

By __________ RC#
Mgr. – Foreman – Supt.

62-4013 (REV. 7-78)

PLEASE COMPLETE AND RETURN TO EMPLOYEE
(EMPLOYEE MUST HAVE COMPLETED CARD TO RETURN TO WORK)

Pacific Gas and Electric Co.: Date __________, 19__

Mr./Ms. __________

Occupation __________ Report # __________

Employed by __________ RC# Division.

Injured at _____ a.m. p.m. on __________, 19__.

☐ Return to full work immediately __________

☐ Modified work until __________

☐ Unable to work until __________

☐ Restrictions or limitations __________

☐ Return Appt. Date: __________ Time: __________

☐ Discharged from treatment __________

Signed __________ MD

PACIFIC GAS AND ELECTRIC COMPANY

PG&E — DIABLO CANYON POWER PLANT
P.O. Box 56 · Avila Beach, California 93424 · (805) 595-7311

R. C. THORNBERRY
PLANT MANAGER

Dear Dr.

Thank you for being one of our panel physicians that treat our employees. Our primary goal is to provide employees who sustain industrial injuries requiring medical attention with prompt, first-class treatment. Your assistance in this endeavor is appreciated.

There is an area of concern to us. While the number of employees that require treatment by a physician has remained stable or in some cases declined, the number of disabling injuries requiring time away from work, i.e., lost time injuries, has dramatically increased.

We believe that some of this time away from work might possibly be avoided if the availability of light (modified) duty or desk-type work were more widely known. Some physicians have stated that in some cases the patient will respond more rapidly to treatment if kept busy in a light-duty capacity. Productive, light-duty assignments are almost always available for employees released for work within the medical restrictions established by the physician.

It our policy to have an injured employee accompanied by a supervisor or other representative on the first doctor's visit. Should there be any question about the availability or type of light duty that can be provided, he or she will be able to answer for us.

Our employees' welfare is our main concern. Should you have any questions about our program, I will be glad to call on you at your convenience.

Sincerely,

R. C. THORNBERRY

RCT.kgb

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 and 2

Safety, Health and Claims Personnel to be Contacted for Reporting of Injuries at Diablo Canyon[1]

Employee Injuries

During working hours:

T. B. Honey, San Francisco,

At any other time:

Report to one of the persons on the following list, trying each in order until one is contacted:

1.	T. B. Honey	Pinole
2.	C. B. Powell	San Francisco
3.	P. S. Benitez	San Rafael
4	B. L. Wade	Larkspur
5.	C. W. Allen	San Francisco
6.	J. A. Glimme	Danville
7.	T. G. Scott	Oakland
8.	A. Thomas	San Francisco
9	J. C. Vocke	Lafayette
10.	W. A. Hutchison	San Carlos
11.	M. C. Dolan	Oakland
12.	A. L. Bechtold	Cupertino
13.	M. W. Johnson	Walnut Creek
14.	R. W. Hall	Richmond
15.	I. M. Crawford	Hercules
16.	R. G. Schumaker	El Granada
17.	R. D. Fagg	San Rafael
18.	P. C. Boettcher	Moraga
19.	H. W. Reynolds	Sunnyvale
20.	B. P. Sadler	Belmont

Non-Employee Injuries

C. O. Schreil, San Luis Obispo, (office)
(office)
(home)

If he cannot be reached, contact one of the following in order of preference:

[1] This listing extracted from Safety, Health, and Claims memo regarding Personnel to be Contacted for Reporting of Accidents, dated 12/8/81.

During working hours:

1. John C. Echols
2. Doug G. Keeler
3. George G. Perry (collection only)

After working hours on Monday through 8:00a.m. on Friday, except holidays:

1.	John C. Echols	Pleasant Hill
2.	Doug G. Keeler	Concord
3.	John C. Vocke	Layfayette
4.	Amos L. Bechtold	Cupertino
5.	William H. Bingaman	Novato
6.	E. Anthony Giudici	San Carlos
7.	J. Alex McCorquodale	San Ramon
8.	Bruce P. Sadler	Belmont
9.	George G. Perry (collection only)	Hayward
10.	Stanley W. Johnston	Fairfield

After 5:00p.m. on Fridays to 8:00a.m. on Mondays and holdays:

Contact the Investigator delegated to stay on call for all emergencies. He may be reached through the System Dispatcher. If he is not available, the Dispatcher will follow the procedures for "After Working Hours".

DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2

EMERGENCY PROCEDURE

TITLE: AREA AND EQUIPMENT DECONTAMINATION

APPROVED: R. C. Thornberry 2-26-82
PLANT MANAGER DATE

SCOPE

This procedure provides guidance on recovering from a radiological emergency that has resulted in contamination of areas or equipment outside of the normal controlled areas onsite.

DISCUSSION

Releases of radioactive material during emergencies may result in contamination of equipment or areas outside of those normally controlled for radiological purposes. Emergency Procedure RB-4, "Access to and Establishment of Controlled Area Under Emergency Conditions," provides guidance on the immediate actions necessary to identify contaminated areas and to control access to these areas. Emergency Procedure RB-4 should be considered a prerequisite to this procedure.

The actions required to decontaminate areas and equipment depend greatly on the material contaminated and the extent and type of contamination present. Certain steps however should occur in any major decontamination effort, as explained in this procedure.

Normal ALARA practices must remain in effect during decontamination efforts, including use of the radiation work permit system for all entries into controlled areas.

RESPONSIBILITY

The Emergency Evaluations and Recovery Coordinator with the assistance of the Emergency Radiological Advisor has the responsibility for implementing the onsite decontamination effort. The Site Emergency Coordinator is responsible for assuring that necessary resources are available to accomplish the decontamination.

Surveys and decontamination work shall only be performed by individuals trained and qualified to perform these functions.

PROCEDURE

1. Perform comprehensive surveys of the affected areas and equipment to determine the amount and type of radioactive contamination present. All survey results must be clearly documented.

2. Based on survey data determine relative levels of contamination on affected equipment and in affected areas. Table 1 provides criteria on acceptable levels of contamination in uncontrolled areas.

   Maps and drawings may be useful in providing an overview of the extent of contamination.

3. Plan the Decontamination Effort

   a. Determine the most appropriate techniques of decontamination. This will be dictated by the materials contaminated, type of contamination present, amount of contamination present and availability of decontamination equipment and materials. In many cases disposal as contaminated waste may be preferable over decontamination.

      Table 2 describes various decontamination techniques.

      Table 3 describes efficiencies of hard surface decontamination techniques.

   b. Determine the chronological sequence in which areas or equipment should be decontaminated. Generally decontamination should proceed from areas of higher contamination to areas of lower contamination. Constraints unique to each situation must be considered however, and certain items (such as those needed in other portions of the recovery effort) may be decontaminated on a priority basis.

   c. Assure that means are available to dispose of radioactive waste generated during the decontamination effort.

   d. Obtain necessary decontamination equipment and supplies.

   e. Develop procedures, instructions, and radiation work permits for those who will perform the decontamination.

   f. Provide training as required on the decontamination techniques that will be employed.

4. Decontaminate the affected equipment and areas as planned.

5. Perform surveys to determine the effectiveness of decontamination efforts.

6. Based on survey results either return equipment and areas to uncontrolled use or plan remedial decontamination (see Step 3). Alternative methods of decontamination should be tried if initial efforts are ineffective. Continue cycles of decontamination and surveys until desired results are achieved.

SUPPORTING PROCEDURES

Radiation Control Procedure G-7, "Radiation Surveys"

RB-4, "Access to and Establishment of Controlled Areas Under Emergency Conditions"

TABLES

1. Acceptable Surface Contamination Levels
2. Methods of Decontamination
3. Hard Surface Decontamination Efficiencies in Percent

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVG[(b)(c)]	MAX[(b)(d)]	REMOVABLE[(b)(e)]
U-nat, U-235, U-238 and associated decay products	5,000 dpm/100cm^2	15,000 dpm/100cm^2	1,000dpm/100cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emiters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000 dpm/100 cm^2	15,000 dpm/100 cm^2	1,000 dpm/100 cm^2

(a) Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits establish for alpha- and beta-gamma-emitting nuclides should apply independently.

Table 1
Acceptable Surface Contamination Levels - Continued

(b) As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

(c) Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

(d) The maximum contamination level applies to an area of not more than 100 cm^2.

(e) The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

TABLE 2

METHODS OF DECONTAMINATION

1. Manual Cleaning

Manual cleaning includes such procedures as wiping, scrubbing, mopping, etc., and in general, is an effective method of removing low or moderate levels of contamination on nonporous or nearly nonporous surfaces. Water or a variety of detergents, solvents, chelating agents, and other chemicals may be used. Manual cleaning usually presents minimal airborne and surface contamination control problems.

2. Mechanical Cleaning

Mechanical cleaning includes such decontamination methods as vacuuming, high-pressure steam and water cleaning, soaking, and ultrasonics. These methods are generally associated with the decontamination of highly contaminated equipment but have application with lower levels of contamination.

a. Vacuuming, Wet or Dry. Vacuuming is generally effective in removing loose particulate contamination and is frequency used as an initial decontamination step preparatory to manual cleaning. Vacuum systems should be properly filtered to prevent the spread of contamination to surrounding areas and to reduce the hazard of airborne contamination. Care should be taken to ensure that the concentration of radioactive material in the vacuum system does not create unusually high radiation exposure rates to personnel and that it does not present a criticality hazard.

b. Jet Cleaning. High-pressure steam and water used alone or mixed with chemicals and detergents are effective in attaining high decontamination factors. Commercial systems using the jet cleaning principle are available. Equipment of this type is ideally suited for remote operation and for cleaning large surface areas. High-pressure jet cleaning has the disadvantage of spreading contamination over a large area and is more effective when used in a cave or cell designed especially for this purpose.

c. Soaking and Spraying. Soaking and spraying are used extensively for decontamination of small and moderate size material and equipment. Both methods make use of chemical solutions and may require support features such as catch tanks, liquid recycle ability, and filtered ventilation systems. Spraying has the

Table 2
Methods of Decontamination (Cont.)

advantage of combining mechanical as well as chemical action; however, in some cases the shape of the object being cleaned prevents effective cleaning action on all surfaces. Soaking provides good access to surfaces but does not provide mechanical action.

d. Ultrasonic Cleaning. Ultrasonic cleaning combines the advantage of chemical action and mechanical energy for cleaning. It is best suited for small components and offers the advantage of remote operation and rapid decontamination of objects with irregular shapes and crevices.

3. Grinding and Abrasive Action

Cleaning procedures employing grinding or abrasive action are effective means of decontaminating metal and concrete surfaces, provided alteration of the surface area of the object being cleaned can be tolerated.

a. Grinding. Grinding of surfaces to remove contamination is usually limited to small objects or isolated spots of contamination where the surface is reasonably smooth. Grinding normally produces a high decontamination factor and is economical. A variety of commercial grinders may be used. Grinding inherently leaves residual contamination on the surface of the object being cleaned and therefore usually requires final cleaning by some other method (vacuuming, wiping, etc.).

b. Abrasive Blasting. Abrasive blasting has a number of advantages over grinding. It is rapid, provides a high DF, is effective on irregular shaped surfaces and can be used for large areas. Abrasive blasting makes use of a large variety of abrasives (sand, shells, glass beads, metals, etc.) with velocity, shape, and size of the abrasive influencing surface-removal characteristics. A prime disadvantage of abrasive blasting is that it usually generates high airborne contamination and spreads surface contamination; however, this can be minimized by wet blasting techniques, vacuum systems, or filtered enclosures.

c. Destructive Decontamination. Destructive decontamination procedures include physical removal of contaminated parts or sections. Generally, little or no effort is made to clean the contaminated parts before disposal as waste. Containment and

Table 2
Methods of Decontamination (Cont.)

other radiological controls associated with destructive cleaning are dependent on contamination levels, the nature of the contaminant, and the physical characteristics of the parts being removed.

Reference: ANSI N13.12. Draft American National Standard, Control of Radioactive Surface Contamination on Materials Equipment and Facilities to be Released for Uncontrolled Use, 1978.

TABLE 3

HARD SURFACE DECONTAMINATION EFFICIENCIES IN PERCENT[(a)]

Material	Vacuum (D+2)[(b)]	Hi-Pressure Water (D+3)[(b)]	Hi-Pressure Wtr.w/Scrub (D+12)[(b)]	Hi-Pressure Wtr. & Detergent (D+4)[(b)]	Hi-Press. Wtr. & Detergent with Scrub (D+5)[(b)]	Sand-blasting (D+9)[(b)]	Steam Cleaning (D+14)[(b)]
Glass	98.95	98.85	97.79	100.00	99.76	100.00	97.86
Stucco	48.00	97.94	95.22	100.00	99.59	100.00	27.00
Painted wood	99.28	98.43	96.77	99.69	99.97	100.00	91.61
Unpainted wood	36.00	85.00	93.18	99.54	95.54	99.90	85.00
Aluminum	89.00	99.45	97.33	99.62	100.00	98.49	84.00
Plate steel	93.04	97.26	94.19	100.00	93.83	99.72	91.46
Asbestos shingles	61.00	99.97	98.91	96.89	99.36	100.00	63.00
Unpainted wood shingles	61.00	97.16	90.49	95.01	57.93	99.82	71.00
Brick	29.99	99.46	99.32	99.14	99.56	99.92	97.50
Tarpaper	55.00	98.66	95.04	95.32	95.83	99.51	52.00
Galvanized roofing	89.00	99.36	97.19	99.73	99.86	100.00	85.00
Highway asphalt	32.00	99.90	96.25	90.82	99.48	99.90	44.00
Highway asphalt (10x10 ft)	72.00	92.45	94.95	98.85	96.34	92.73	22.00
Steel asphalt	71.00	98.67	90.00	100.00	99.72	99.61	84.00
Steel asphalt (10x10 ft)	64.00	90.00	82.00	96.31	97.54	90.42	48.00
Steel trowel concrete	74.00	98.94	--	96.91	99.53	100.00	--
Steel trowel concrete (10x10 ft)	--	73.00	97.34	--	99.58	98.96	27.00
Wood float concrete	--	98.00	92.03	100.00	97.47	100.00	65.00
Wood float concrete (10x10 ft)	56.00	97.84	--	98.09	98.28	98.78	85.00
Avg. of all surfaces	65.40	96.12	94.59	98.61	98.64	98.83	67.80

(a) Decontamination factor (DF) = 100/[100 - decontamination efficiency (%)]

(b) (D+n) = number of days between contamination and decontamination

WASH 1400, Appendix VI, October 1975, "Reactor Safety Study."

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