PGandE Letter No.: DCL-84-090

ATTACHMENT 1

Updates Included In This Submittal

DIABLO CANYON EMERGENCY PLAN
Implementing Procedures

Volume 3A

EP OP-0, Revision 5
EP OP-1, Revision 7
EP OP-8, On-The-Spot-Change
EP M-9, Revision 1

Volume 3B

EP RB-7, Revision 3
EP RB-9, Revision 2

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ATTACHMENT 2

Location of Proprietary/Privacy Information

Procedure:

RB-7, pages 32 and 33 of 33

M-9, pages 2 and 5; attachment, "Hazardous Waste Spill Notification Requirements"; attachment, "Information Required for Initial Reporting of Hazardous Waste Release (Check List)."

CURRENT

EMERGENCY PLAN

IMPLEMENTING PROCEDURES

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03/01/84

IMPORTANT TO SAFETY

TITLE: REACTOR TRIP WITH SAFETY INJECTION

APPROVED: R. C. Thornberry 2-8-84
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. This procedure and changes thereto require PSRC review.

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

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[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 Generator 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.	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 B 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.

NOTE

On spurious SI recovery, if RCS pressure decreased below the shutoff head of the SI pumps (1520 PSIG) during the transient, perform leak test (STP V-5) within 24 hours following the ECCS actuation on those valves listed in Table 3.4-1 of Tech Spec 4.4.6.2.2 which actuated or if there was flow through the valve.

AND k. Pressurizer water level is greater than 22%.

ACTION	COMMENTS
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	COMMENTS
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.	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.
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,	

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

TITLE: REACTOR TRIP WITH SAFETY INJECTION

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. IF 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 AND the pressurizer water level remains above 10%, AND the reactor coolant indicated subcooling is greater than 35°F, then consider the event 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. CAUTION: Do not reset the reactor trip breakers until authorized by the Plant Superintendent.

SPURIOUS SI SIGNAL RECOVERY

ACTION	COMMENT
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.

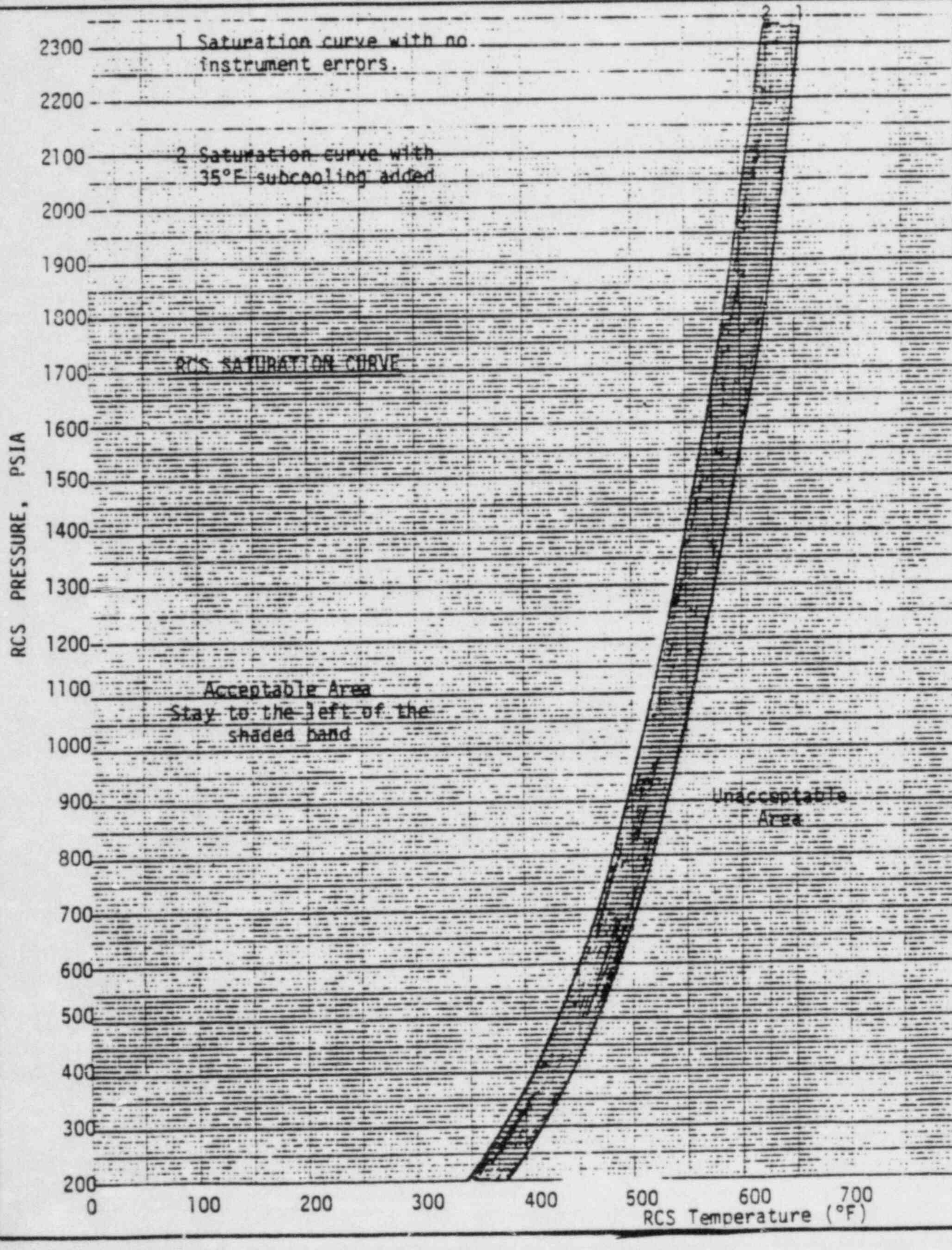
2) Verify the GFFD countrate returns on scale and stabilizes at a value below the post SI countrate on the recorder.

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	NUMBER EP OP-0 REVISION 5 DATE 2/2/84 PAGE 16 OF 20
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 are 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. Monitor the thermocouple readout on PAMS 3 and 4. If 5 or more 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 readings that exceed 1200°F, discontinue this appendix but continue to monitor the thermocouple readings.

4. 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 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.
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.

IMPORTANT TO SAFETY

TITLE: EMERGENCY OPERATING PROCEDURE
LOSS OF COOLANT ACCIDENT

APPROVED: R. C. Thornberry 2-8-84
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.

This procedure and changes thereto requires PSRC review.

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 recirculation of spilled reactor coolant, injected water and containment spray drainage.

3. In the case of the small break LOCA, to reduce plant conditions to 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 <u>the</u> 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 items 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

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

   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:

COMMENTS

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.

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. 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.	e. Charging at a rate greater than normal might be required in order to maintain PZR pressure and level.

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 cool-down 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 sub-cooling 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 after:	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 cent·'s 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. DO NOT 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 until the RWST low level RHR pump trip occurs. (Trip the RHR pumps at 33% RWST level if the automatic trip fails, then perform 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	COMMENTS
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 Spray 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. <u>Verify ECCS flows appoximately every 15 minutes during the cold leg recirculation phase</u>.	
During this period, make the following motor operated valves available by closing their 480V <u>breakers</u> at the vital load centers.	33. <u>DO NOT</u> make any valve changes after making the valves available.
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.	

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 if this has not been performed.	
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. 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 safety 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 flo 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: a. Open idle CCW heat exchanger outlet valve (FCV-430 or FCV-431). b. Open saltwater inlet valve on idle heat exchanger FCV-602 or FCV-603). c. Open stop valves of idle heat exchanger in the water box vent lines to air removal tank (el. 119'). d. Verify CCW heat exchanger saltwater outlet valve (SW-65 or SW-66) throttled at 56° to maintain water box volume. e. Verify that operating standby ASW pump discharge valve is open (SW-54 or SW-55). f. Close motor operated pump discharge crosstie valves (FCV-495 and FCV-496). g. Verify approximately equal motor current on each ASW pump motor and equal flow through each heat exchange.	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: a. Verify that the idle CCW heat exchanger outlet valve (FCV-430 or FCV-431) is open. b. Open or verify open FCV-355 Header "C" supply. c. Close pump suction header crosstie valve (CCW-4) between headers "B" and "C". d. Close CCW Pump 1 discharge valve (CCW-18) to header "A". e. Close CCW Pump 2 discharge valve (CCW-16) to header "B". f. Close CCW Pump 3 discharge valve (CCW-17) to header "B". g. Isolate CCW header "C" from header "B" by closing valve CCW-24.	25. Same note as 24 above.

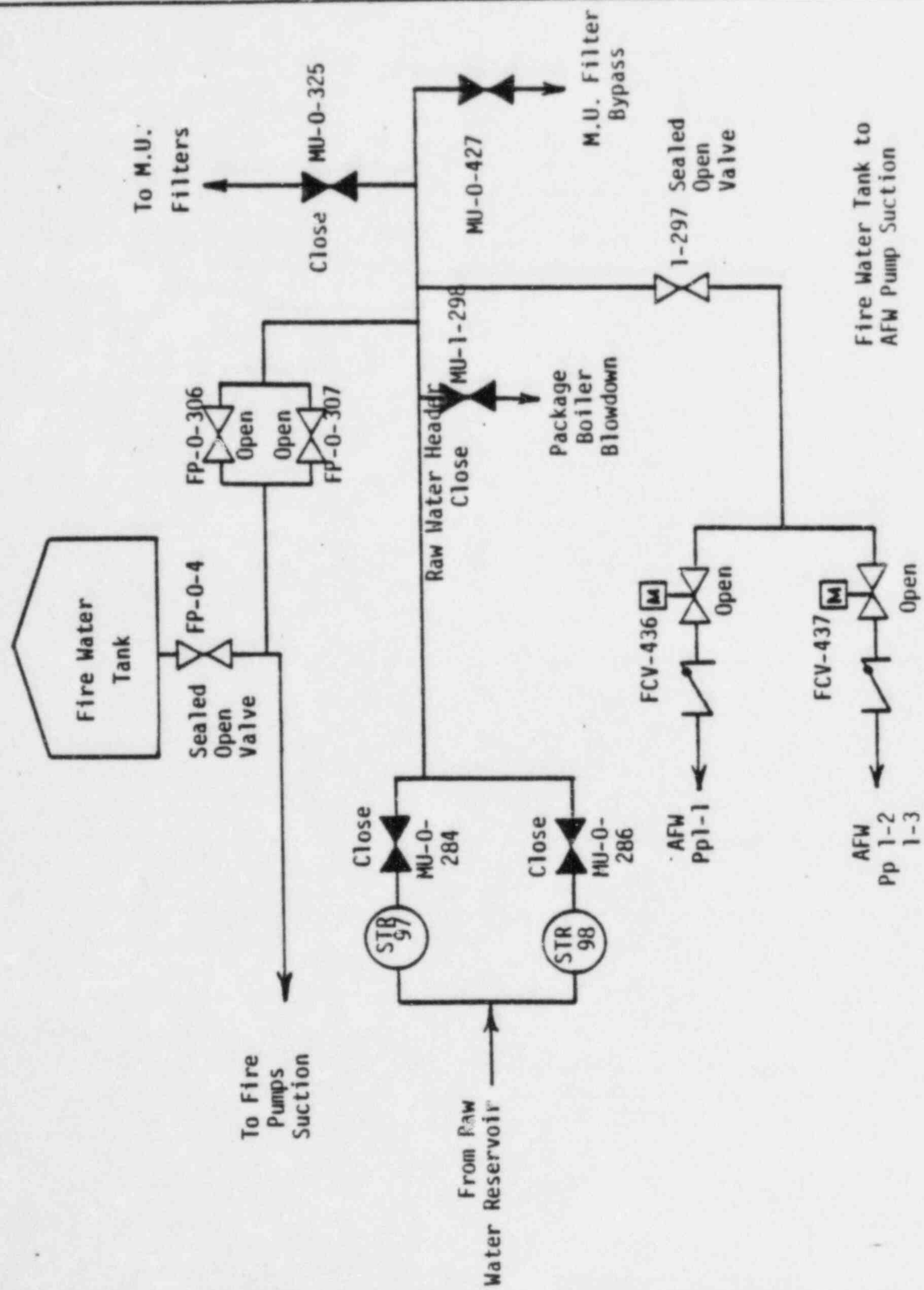
APPENDIX C

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.	


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
FP-0-306
Open
Open
FP-0-307
MU-1-298
Raw Water Header Close
Package Boiler Blowdown
Fire Water Tank
FP-0-4
Sealed Open Valve
FCV-436
Open
FCV-437
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

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 set-point, trip all safe-guards 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 8703 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 preve[illegible] to the cold [illegible] the chargi[illegible] 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. Monitor the thermocouple readout on PAMS 3 and 4. If 5 or more 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 readings that exceed 1200°F, discontinue this appendix but continue to monitor the thermocouple readings.

4. 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. Longitudi[illegible]	________	x 0.01 =	________	g
b. Vertical	________	x 0.01 =	________	g
c. Transverse	________	x 0.01 =	________	g

BY ______________________ TIME ______________

3. CONTAINMENT PRESSURE

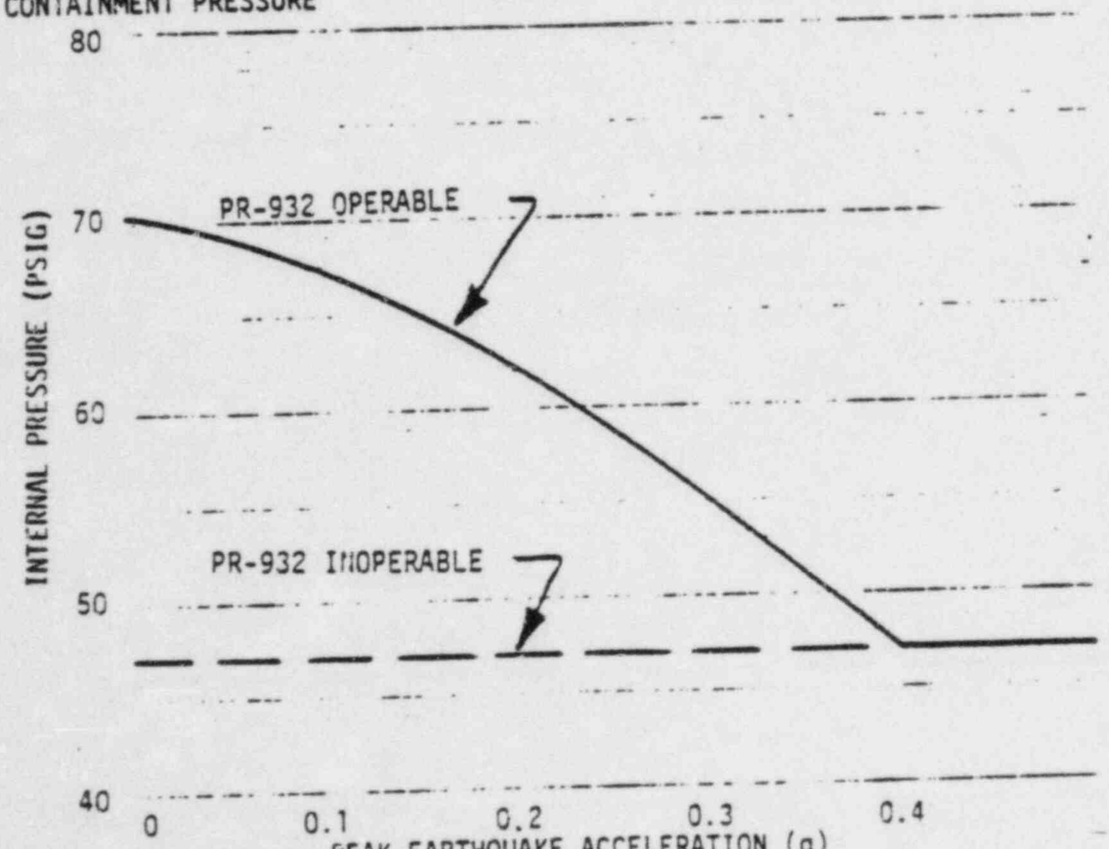


Figure 1
Internal Pressure Capability of the Containment
Versus Earthquake Acceleration

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 <u>lowest reading</u> on <u>one</u> 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), <u>or</u> 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

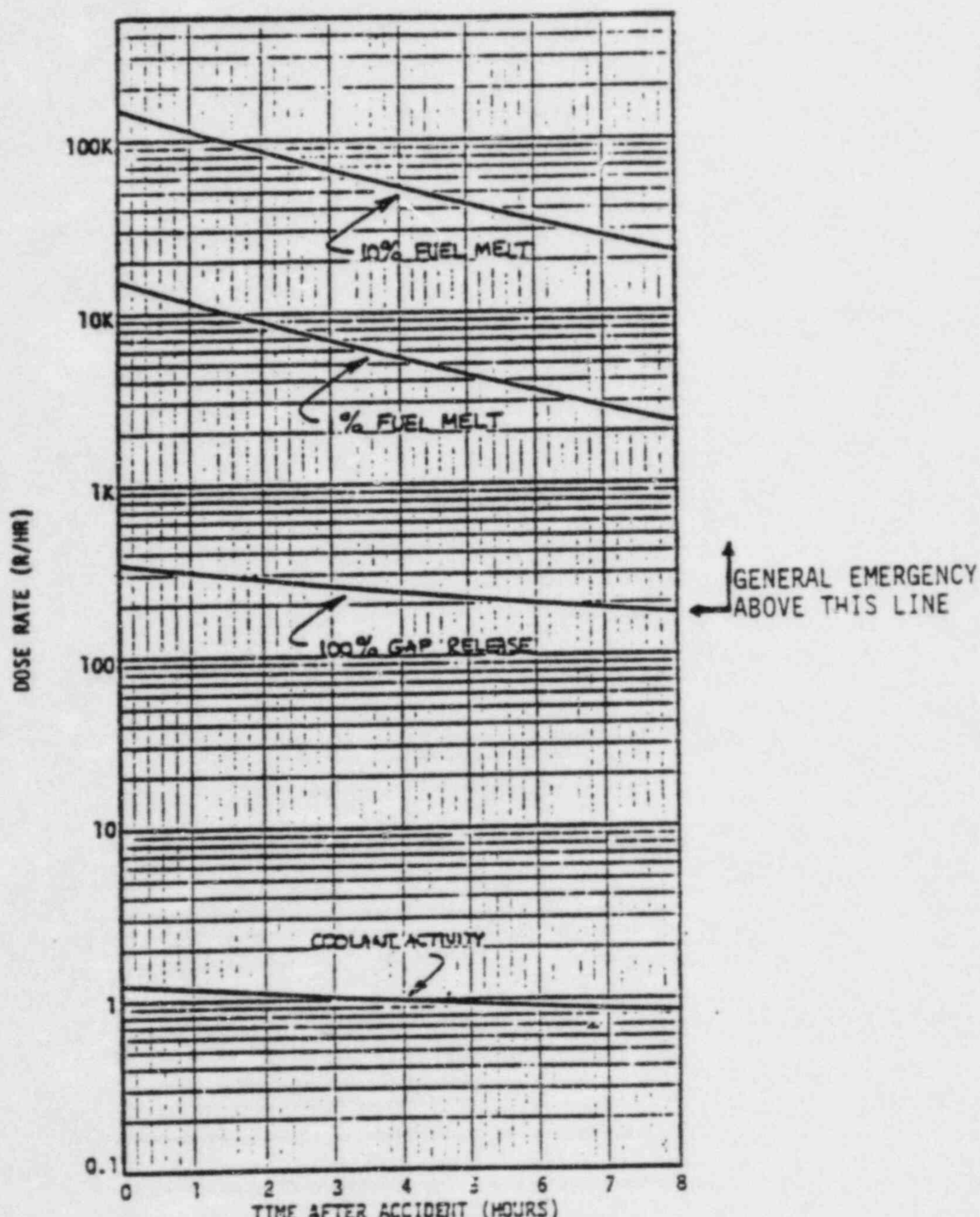


FIGURE 1

DOSE RATE INSIDE CONTAINMENT FOLLOWING LOCA

WORKSHEET 3, APPENDIX H

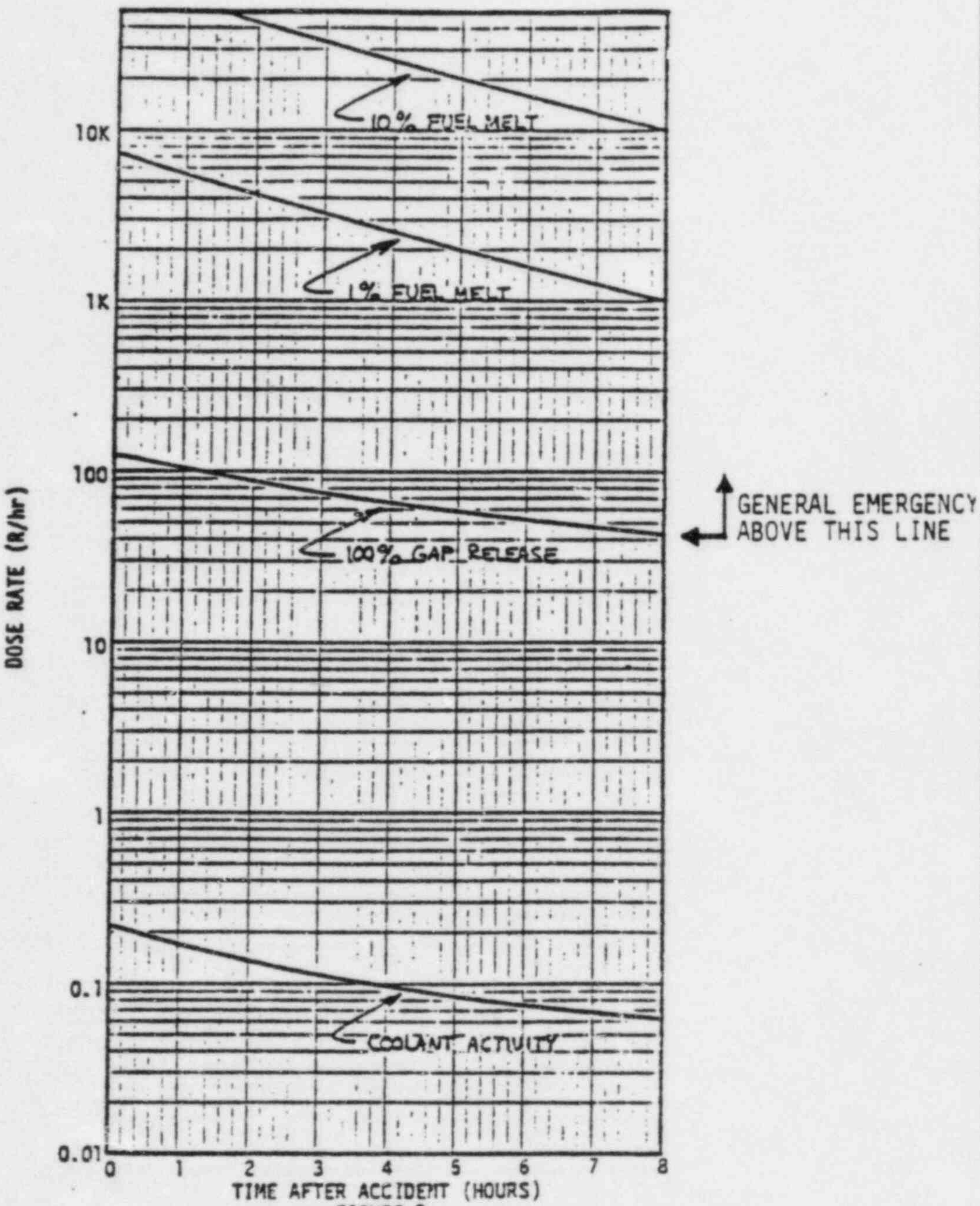


FIGURE 2

DOSE RATE OUTSIDE EQUIPMENT HATCH LINER FOLLOWING LOCA

WORKSHEET 3, APPENDIX H

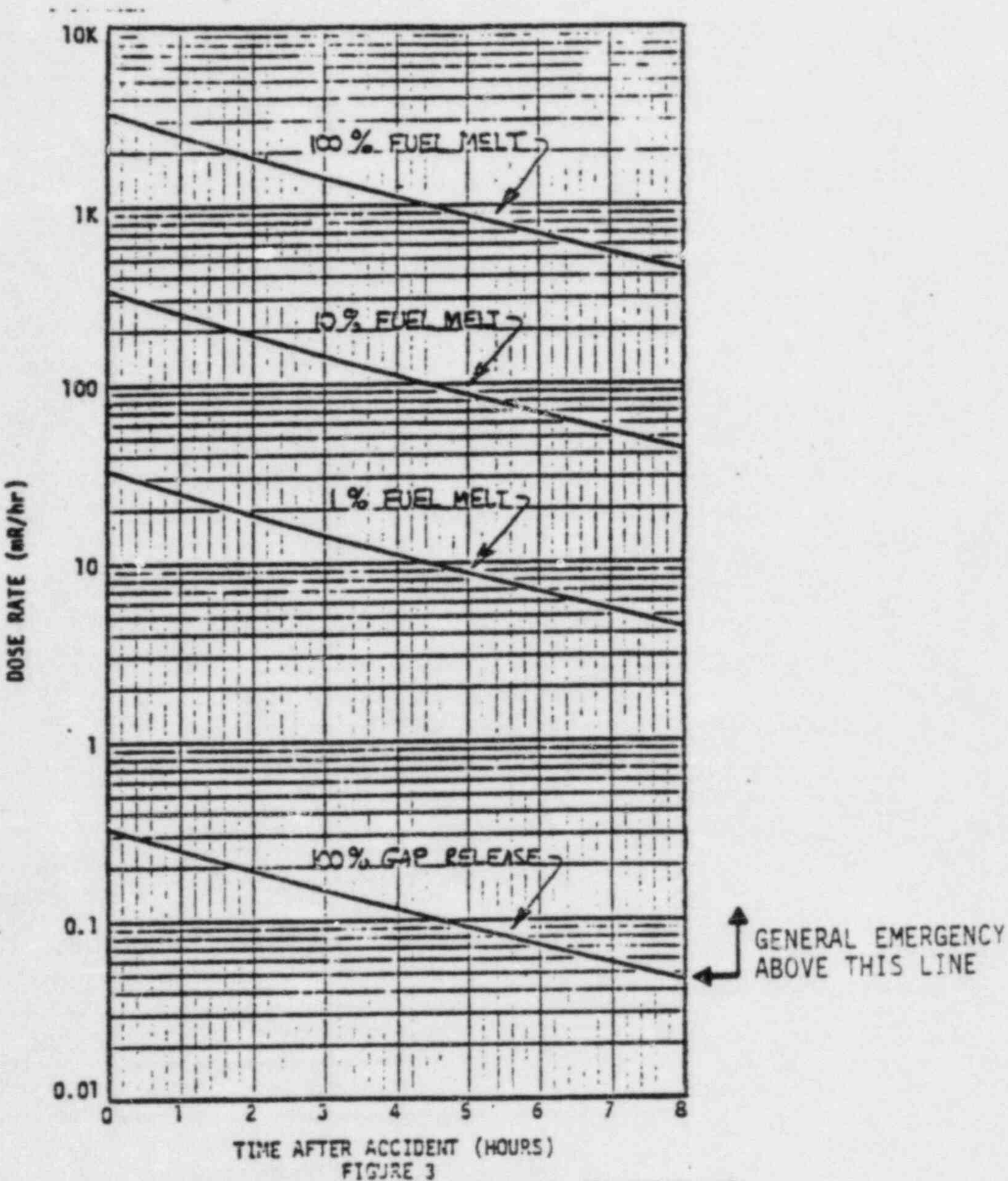
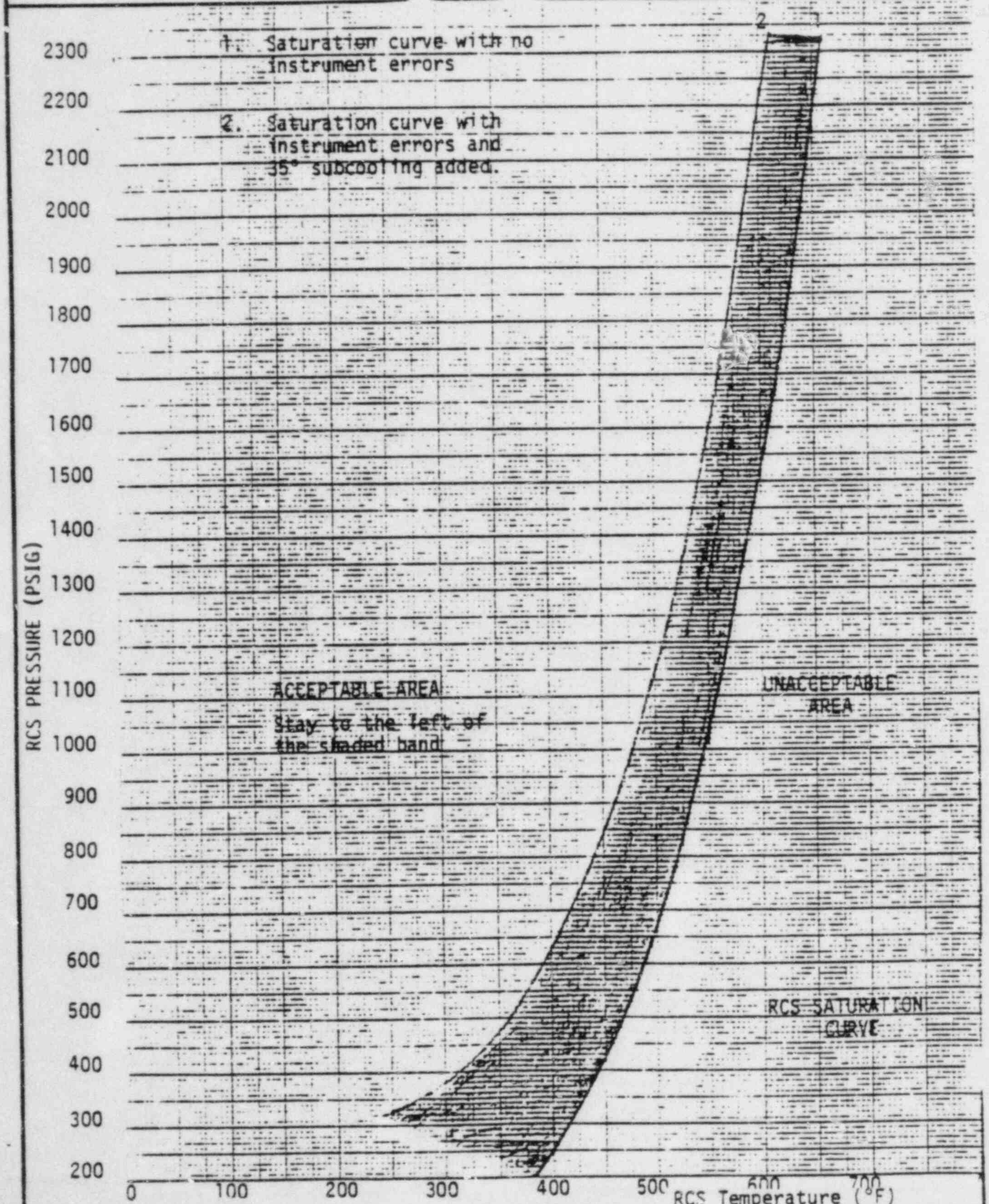


FIGURE 3
DOSE RATE OUTSIDE OF CONTAINMENT SHIELD WALL FOLLOWING LOCA


1. Saturation curve with no instrument errors
2. Saturation curve with instrument errors and 35° subcooling added.
2
1
2300
2200
2100
2000
1900
1800
1700
1600
1500
1400
1300
1200
1100
1000
900
800
700
600
500
400
300
200
RCS PRESSURE (PSIG)
ACCEPTABLE AREA
Stay to the left of the shaded band
UNACCEPTABLE AREA
RCS SATURATION CURVE
0
100
200
300
400
500
600
700
RCS Temperature (°F)

PG&E
Nuclear Plant Operations
69-10700 10/83

DIABLO CANYON POWER PLANT
PROCEDURE ON-THE-SPOT CHANGE

ORIGINATOR

Procedure No. EP OP-B Rev. 7 Unit No. 1 ☐ 2 ☐ 1 & 2 ☒

Title Control Room Inaccessibility

Type of Change: ☒ PERMANENT (green) ☐ TEMPORARY (yellow); Expiration Date

Requesting Department Operations Originator J. Bard

Proposed Change: (Does this alter the intent of original procedure? ☐ Yes ☒ No)
(Does it constitute an unreviewed safety/environmental question? ☐ YES ☒ NO)

Page 4, item 1.i revise to read:
"ASW pump 1-2"

Page 5, item 2.c.2 revise to read:
"ASW pump 1-1"

Reason for Change:
Normal positions of the ASW pump control transfer cutout switches have been changed to ensure the operability of ASW pump 1-1 for a fire at the HSD panel. (DCI-83-TN-P0214)

Authorizations: J. Bard (Plant Management Staff) [signature] (Plant Management Staff w/SRO License) 2/20/84 Date*

Immediate distribution to the Control Room and/or affected work areas required? ☐ YES ☒ NO

Initial Distribution By:

Distributed to: ☐ Control Room ☐ Others

DOCUMENT CONTROL

Date Received by Document Control 2-21-84

PSRC Review and Plant Manager's approval no later than 3-5-84 Date above *plus 14 days

PSRC POST CHANGE REVIEW

Review Date

PSRC recommends approval ☐ Yes ☐ No

Plant Manager's Approval ☐ N/A

Meeting Number ☐☐-☐☐☐

[ORIGINAT]ING DEPARTMENT

Follow-up To Rejected On-the-Spot Change ☐ Additional Information ☐

Action Taken/Remarks:

DISTRIBUTION: ☐ Same as Original Procedure Distribution ☐ Others

Please see additional sheets ☐

TITLE EMERGENCY PROCEDURE
HAZARDOUS WASTE MANAGEMENT CONTINGENCY PLAN

APPROVED: R. C. Thornberry 2-15-84
PLANT MANAGER DATE

IMPORTANT TO
ENVIRONMENTAL QUALITY

SCOPE

This procedure describes the responsibilities and the actions which are taken to minimize hazards to human health or to the environment from any unplanned release of hazardous waste or its constituents to the air, soil, or surface water.

GENERAL

The Hazardous Waste Management Contingency Plan (Contingency Plan) meets the requirements of Title 22 of the California Administrative Code.

The Spill Prevention Control and Countermeasure Plan for Diablo Canyon Power Plant (SPCC Plan) which covers the handling of oil spills, will be used in conjunction with the Contingency Plan, if applicable.

RESPONSIBILITY

The Shift Foreman (Site Emergency Coordinator) will assure that the provisions of the Contingency Plan are carried out in a manner which meets the requirements of the Operation Plan for a Hazardous Waste Facility.

Any spill or leakage of hazardous waste will be reported immediately to the Shift Foreman on duty by the first employee having knowledge of the situation.

The Chemistry and Radiation Protection (C&RP) Department will provide technical assistance to the Shift Foreman in assessing possible hazards due to the unplanned release of hazardous waste, and will assure proper disposal of hazardous waste.

NOTE: The DCPP Emergency Plan provides for a Site Emergency Coordinator to be available at the DCPP during all three shifts of operation (continuous 24-hour per day, 365 days per year). The on-duty Shift Foreman [Extension 1234] is the designated Site Emergency Coordinator. If the emergency is extensive or other conditions warrant, the plant Manager [(805) 595-7351 Extension 3350] or his designee may assume the duties of the Site Emergency Coordinator.

PROCEDURE

1. IMMEDIATE ACTIONS

In the event of an imminent or actual emergency situation involving the release of hazardous waste, the Shift Foreman will take the following steps as expeditiously as possible;

a) Activate the site emergency signal as appropriate, to clear the affected area and alert personnel that an emergency has occurred.

b) Determine if any personnel are injured.

c) Immediately dispatch a operator to the scene to determine the type (from NFPA placard if release is from a tank), the exact source, the amount of the spill or leak, the extent of the area affected, and if the hazardous waste:

1) Presents an immediate threat to life or health;
2) Extends beyond the property line of the power plant;
3) Could enter a nearby body of water;
4) Could create a fire or explosion hazard;
5) Could contaminate the atmosphere;
6) Could interfere with power plant operation;
7) Could mix with other materials and create an additional hazard.

e) Immediately, contact the on-shift C&RP Technician for technical assistance.

f) With the assistance of the C&RP Technician, assess possible hazards to human health or to the environment that may result from the release. Both the direct effect of the hazardous waste and indirect effects, which might result from the actions taken to correct the situation, should be considered.

g) Instruct the Control Operator or other individuals to make the required telephone calls to request assistance and to notify other Company personnel, public agencies and outside contractors. The telephone numbers of Company personnel and governmental agencies are listed in Attachment 1. The information to be reported is outlined in Attachment 1.

h) Determine the source and cause of the release and prevent further release by taking appropriate action, including the following:

1) Securing the valves;
2) Directing the waste to holding tank;
3) Collecting the waste in containers; and
4) Stopping the operation that produced the waste.

i) Initiate actions and physical means for containing and isolating the affected areas in order to prevent further spread of the waste or the mixing of the waste with incompatible materials. Barricades, barrier tape, warning signs, oil spill absorbent, and other available means shall be used as required.

j) Record in the Shift Foreman's Log Book, the time, the date, and the details of any incident that required the use of the Contingency Plan. The record shall be sufficiently detailed to serve as the basis for the required written report to the California State Department of Health Services.

2. After the release of hazardous waste has been brought under control, the Shift Foreman will initiate actions to:

a) Inspect valves, pipes, and other waste handling equipment for leaks, pressure buildup, gas generation, or rupture, wherever this is appropriate.

b) Collect and retain the recovered hazardous waste and contaminated soil, water, or other material in compliance with the regulations, taking care that no incompatible wastes are mixed.

3. The C&RP Department will:

   a) Arrange for the shipment of all hazardous waste and contaminated soil, water, or other material recovered during the emergency to an EPA-approved disposal site via a registered waste hauler.

   b) Assure that no materials that may be incompatible with the released material is brought into the affected area until cleanup procedures have been completed.

   c) Have all emergency equipment cleaned, restored to proper condition, and returned to its location. The overall clean-up procedure shall not be considered complete until this has been done.

4. Reports to California State Department of Health Services

   a) Telephone Report

   Within 24 hours of the occurrence, the Plant Manager or his designee will notify the California State Department of Health Services by telephone of all accidents involving hazardous wastes which resulted in, or could have resulted in, a hazard to public health and safety, or the environment.

   After the clean-up has been completed, the Plant Manager or his designee will notify the California State Department of Health Services. If applicable, other individuals or agencies previously notified of the release will be notified that the clean-up has been completed.

   b) Written Report

   Within 30 days after an incident that requires implementation of the Contingency Plan, a written report on the incident will be prepared by the C&RP Department for submission to the California State Department of Health Services.

NOTE: The DCPP Emergency Plan provides for a Site Emergency Coordinator to be available at the DCPP during all three shifts of operation (continuous 24-hour per day, 365 days per year). The on-duty Shift Foreman [Extension [redacted] is the designated Site Emergency Coordinator. If the emergency is extensive or other conditions warrant, the plant Manager [redacted] Extension [redacted] his designee may assume the [redacted] Site Emergency Coordinator.

PROCEDURE

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In the event of an imminent or actual emergency situation involving the release of hazardous waste, the Shift Foreman will take the following steps as expeditiously as possible;

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   1) Presents an immediate threat to life or health;
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   4) Could create a fire or explosion hazard;
   5) Could contaminate the atmosphere;
   6) Could interfere with power plant operation;
   7) Could mix with other materials and create an additional hazard.

e) Immediately, contact the on-shift C&RP Technician for technical assistance.

f) With the assistance of the C&RP Technician, assess possible hazards to human health or to the environment that may result from the release. Both the direct effect of the hazardous waste and indirect effects, which might result from the actions taken to correct the situation, should be considered.

REQUIREMENTS

1. Arrangements with Authorities

Arrangements have been made with the following authorities to assist with emergency situations at DCPP:

a) U. S. Coast Guard
b) State Department of Parks and Recreation
c) San Luis Ambulance Service
d) Aris Helicopter Service
e) French Hospital
f) California Department of Forestry
g) Fire Department of Local Cities (under CDF di[redacted]
h) California Highway Patrol
i) Department of Fish and Game
Answering Service
Nights & Holidays, Warden Drew Brandy, (home)

2. Emergency Equipment

DCPP is equipped with the following emergency equipment:

a) Telephones are distributed throughout the plant within easy reach of all employees. Emergency instructions are posted by the phones.

b) An alarm and communication system is established at the site to alert employees of emergency situations and to summon emergency assistance from the county and state emergency response teams.

c) Safety showers and eyewash stations in working condition and available for immediate use.

d) Eyewash bottles are provided for the more remote waste areas.

e) Industrial type first aid kits and burn kits are maintained at various locations in the plant.

f) Various types of portable fire extinguishers and automatic fire control systems, e.g.,

. 20-lb dry chemical
. 17-lb Halon 1211
. 15-lb CO_2
. 2½ gallon water
. Automatic sprinkler and deluge system
. Automatic Halon (1301) spray
. Automatic Carbon Dioxide

All fire fighting systems are tested on regular basis. Test and maintenance frequencies are described in the Surveillance Test Procedures Manual.

g) Protective clothing and equipment approved by the National Institute of Occupational Safety and Health (NIOSH):

1) MSA (trademark) Respirators as follows:
25 - ½ face filter respirators, located at the plant's maintenance tool room;
75 - full face filter respirators, located at the plant's Access Control;
30 - in line full face respirators, located at the plant's Access Control; and
15 - full face powered filter respirators, located at the plant's Access Control.

Appropriate cartridges for filter face respirators are used for chlorine gas, organic vapors, acid fumes, ammonia, and particulates.

2) 90 MSA self-contained breathing apparatus including self-contained pressure demand units and 50 spare bottles, located at:

. Control Room (inside the Auxiliary Building)
. 2 Fire Lockers (inside the Auxiliary Building)
. Technical Support Center (on the plant site)
. Radiological Access Control (inside the Auxiliary Building)
. General Construction Warehouse (south of the plant)

3) Goggles: Faceshields are available to any employee handling hazardous substances.

4) Boots:
100 pair latex overboots; duration of use - 3 years
500 pair rubber overshoes; duration of use - 3 years
Several thousand plastic, tearaway shoe covers; duration of use: used once then thrown away.

5) Coveralls: Made of cotton, fire retardant; duration of use - 3 years.

6) Gloves:
Latex gloves; duration of use - 2 months.
Heavy duty gauntlets; duration of use - 1 year.

7) Material safety data sheets on chemicals used at the plant.

3. Evacuation Plan

Diablo Canyon Power Plant has developed an extensive Emergency Plan which includes procedures to perform an evacuation of nonessential site personnel. The Site Emergency Coordinator (Shift Foreman) will assess the need or the potential need for personnel evacuation. However, considering the locations of and the waste stored in the hazardous waste areas at the plant, it is unlikely that evacuation will be necessary.

4. Amendment of the Contingency Plan

The Contingency Plan shall be reviewed, and revised under any one or more of the following conditions:

1) the applicable regulations were revised
2) the plan failed in an emergency
3) the list of the emergency coordinators has been changed; or
4) the list of the emergency equipment has been changed.

5. Copies of the Contingency Plan

The Contingency Plan and all revisions to the plan will be kept on file by the Plant Manager at Diablo Canyon Power Plant.

Copies of the Contingency Plan will be incorporated into the Hazardous Waste Operation Plan for Diablo Canyon Power Plant for use by the Shift Foremen and other plant personnel.

Copies of the Contingency Plan will be provided to other Company personnel and to outside agencies or organizations who may be called upon to provide emergency services.

A copy of the Contingency Plan will be submitted to the California State Department of Health Services for approval.

REFERENCE

1. DCPP Interim Status Document, April 1981
2. DCPP Emergency Procedure G.4
3. DCPP Emergency Procedure G.5

ATTACHMENTS

1. Hazardous Waste Spill Notification Requirements
2. Information Required For Initial Reporting Of Hazardous Waste Release (Check List)

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

TITLE: HAZARDOUS WASTE SPILL NOTIFICATION REQUIREMENTS

In the event of a spill or leak of hazardous waste which could threaten human health or the environment, the following must be notified as soon as possible.

Government Agencies

[redacted]

a) National Response Center

b) California State Office of Emergency Services (if spill affects area outside plant property)

c) U.S. Coast Guard

d) California Department of Health Servic[redacted]
- Fresno Regional Office

e) California Regional Water Quality Control Board - Central Coast Region

PGandE Personnel

Office Phone

[redacted]

Plant Manager

Supervising Chemical and Radiation Protection Engineer

Senior Chemical and Radiation Protection Engineer

Chemical and Radiation Protection Engineer

Supervising Nuclear Generation Engineer, Personnel and Environmental Safety

Law Department

DC0227 9I

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

TITLE: INFORMATION REQUIRED FOR INITIAL REPORTING OF HAZARDOUS WASTE RELEASE (CHECK LIST)

When a governmental agency is notified by telephone of a release of hazardous waste which threatens human health or the environment, the following information should be available.

1) Name of reporter: [redacted]
   Telephone number: [redacted]

2) Name and address of facility:
   Pacific Gas and Electric Company
   Diablo Canyon Power Plant
   Avila Beach
   San Luis Obispo, CA

3) Date and time of incident: ____________________
   Type of incident
   a) Spill of hazardous waste
   b) Leakage of hazardous waste
   c) Fire
   d) Explosion
   e) Other: ____________________

4) Name of materials involved (to the extent known). Include approximate pH and toxic concentration of material.
   Quantity of materials. ______________ (to the extent known):

5) Extent of injuries, if any: ____________________

6) Possible hazards to human health, or the environment, outside the facility.

   Areas affected:
   a) Inside plant property
      - Notify only Company personnel unless directed otherwise
   b) Outside plant property
   c) Diablo Cove
   d) Diablo Creek
   e) Contamination of atmosphere

CURRENT

EMERGENCY PLAN

IMPLEMENTING PROCEDURES

TABLE OF CONTENTS

Volume 3B

03/01/84

0212d/0007K

TITLE: EMERGENCY PROCEDURE
EMERGENCY ON-SITE RADIOLOGICAL
ENVIRONMENTAL MONITORING

IMPORTANT TO SAFETY

APPROVED: R. C. Thornberry 2-8-84
PLANT MANAGER DATE

SCOPE

This procedure describes an initial emergency radiological environmental monitoring program to determine gamma and beta dose rates and air particulate and radioiodine levels due to an accidental release of gaseous radionuclides. This data will be used to make initial assessments concerning the magnitude of the accident and decisions concerning evacuation of nonessential site personnel. This program can be undertaken early in the accident assessment process by a single team working on or near the site. This procedure and changes thereto requires PSRC review.

GENERAL INSTRUCTIONS

1. The on-site radiological environmental monitoring team(s) are under the direction of the Emergency Radiological Advisor (ERA). If the ERA has not arrived on-site, the Emergency Evaluations and Recovery Coordinator will direct the monitoring team(s).

2. As a minimum, the on-site radiological environmental monitoring team(s) will be comprised of two on-shift personnel; a Chemical and Radiation Protection Technician (C&RP) (team leader) and an Auxiliary Operator (assigned by the Emergency Evaluations and Recovery Coordinator).

3. The on-site monitoring team(s) will maintain communication with the Control Room and/or TSC through the use of hand-held radios. Use the Health Physics channel. ("T" set on "1", "F" set on 5 for repeater, 6 for local).

4. The team should have the following equipment:

   a. Emergency kit or equipment equivalent to it's contents. The emergency kits are located in the exit area of the Operational Security Center.

   b. Hand-held radio. These are located in the Operational Security Building and at Access Control.

c. Vehicular transport

d. Life jacket, if monitoring is to be done on the breakwater. Contact the Shift Foreman for location of the jackets.

5. When traveling from one location to another, always have a detector on so that if a portion of the plume is traversed, it will be recognized immediately. If traveling in a vehicle, hold a detector out of the window. If any location is identified as being above background, measurements should be taken and data recorded.

6. Prior to on-site out-of-plant monitoring, readings from the fixed monitors, PICs, should be obtained as well as information from any preliminary surveys conducted in-plant. This will provide some insight as to what on-site out-of-plant situations might be encountered.

7. Protective clothing should be worn if loose surface contamination levels above unconditional release are expected. The amount of and the need for protective clothing must be balanced against other controls (such as leaving the area). Here are some criteria to be used in determining when and how much protective clothing is worn.

a. If contamination surveys (e.g. smears, direct ground or surface surveys, probe held out the car window, facing down, increase in background) indicate $\geq$1000 dpm/100cm^2 but <10,000 dpm/100cm^2 the minimum protective clothing is hand and feet protection.

b. If surveys indicate activity $\geq$10,000 dpm/100cm^2 then a minimum of a full set of protective clothing is worn.

8. Respirators should be work if surface contamination levels of $\geq$100,000 dpm/100cm^2 is found or an air particulate sample indicates an concentration of $5x10^{-8}\mu Ci/ml$ for unknown beta-gamma emitters. The following rules of thumb can be used to determine the number of counts per minute (as a function of sample volume) that would give the above concentration.

a. Using an HP-210 or HP-260 probe, the ratio of cpm to sample volume to give $5x10^{-8}\mu Ci/ml$ = 500 cpm/ft^3

b. Using an HP-240 probe (window open) the ratio of cpm to sample volume to give $1x10^{-8}\mu Ci/ml$ = 50 cpm/ft^3

For example, if a sample volume of $10ft^3$ on a particulate filter was taken, a 5000 cpm reading using an HP-210 probe would be equivalent to $5x10^{-8}\mu Ci/ml$.

NOTE: The presence of radon products can cause erroneous readings. Be sure to count filter with and without a sheet of paper between it and the probe. If the count rate decreases by 50% with the paper then the rules of thumb need to be adjusted a corresponding amount.

COMMUNICATIONS

1. For ease of communication with the radios, the field monitoring teams will be given call names using the International (ICAO) Phonetic Alphabet:

A - Alpha	E - Echo	I - India
B - Bravo	F - Foxtrot	J - Juliett
C - Charlie	G - Golf	K - Kilo
D - Delta	H - Hotel	L - Lima

2. Radio Technique

   a. Hold the radio upright, directly in front of the mouth with antenna oriented 90° from direction of receiving station (e.g., TSC). Normally, then the antenna is also upright.

   b. Before transmitting, make certain that someone else is not already transmitting on the frequency.

   c. After pushing transmit button, wait two seconds to allow automatic radio encoding to occur.

   d. Begin all communications using the following example:

      "TSC, this is Alpha team. Do you read me?"

   e. Close all communications using the following example:

      "This is Alpha team. Over."

3. Each monitoring team shall contact the TSC under the following conditions:

a. Prior to beginning the monitoring program, in order to obtain initial instructions.

b. Upon completion of monitoring at each location

OR

c. At least once per hour.

PROCEDURE

1. Determine Affected Downwind Sectors

   a. Receive information from the Emergency Evaluations and Recovery Coordinator, or:

   b. Determine wind direction from meteorological computer output or other suitable means. Wind directions are given as the direction from which the wind is blowing.

   c. Figure 1, "Onsite Assembly and Monitoring Location," shows a site map indicating sixteen 22.5° sectors. In general, the areas which may be affected include the 22.5° sector directly downwind plus the 22.5° sector on either side of the downwind sector.

2. Determine if Personnel Assembly Areas are Affected

   Figure 1 also shows the various personnel assembly areas. If any occupied assembly areas are in an affected sector(s), make the measurements specified in Step 5 below at the affected assembly area(s).

3. Check Site Monitoring Locations

   After checking the personnel assembly areas, make the measurements specified in Step 5 below in each of the three affected sectors. Measurements should be made at as many of the locations shown on Figure 1 as it is feasible to reach. Table 1 provides a description of emergency on-site monitoring locations.

   a. Take the first measurement at the location which best approximates the direction downwind location, then take measurements at the adjacent locations on either side of this locations.

b. It may be necessary to make measurements at additional locations, depending upon meteorological conditions.

NOTE: The on-site monitoring locations shown on Figure 1 are identified by a white pole, approximately six feet high, topped with a red circular disc.

CAUTION: Do not attempt to make measurements on the breakwaters if there is a hazard from high seas. At any time measurements are made on the breakwater, the Liaison Coordinator should be informed immediately before and after entry onto the breakwater. Life jackets should be worn, if possible, whenever a person goes onto the breakwater.

4. Initiate Population Center Monitoring

a. If the monitoring performed in Step 2 or 3 shows levels of 3 mr/hr or 9,000 cpm, then a monitoring team should be dispatched to the nearest downwind population center as soon as possible. Protective measures listed in the General Instructions should be reviewed by the monitoring team.

b. Figure 2, "Emergency Off-site Monitoring Locations", shows the standard off-site environmental monitoring locations. Table 2, "Description of Emergency Off-site Monitoring Locations", describes each location; and Table 3, "Preferred Monitoring Locations for Initial Population Center Monitoring Locations", recommends those which should be considered for initial population center monitoring as a function of wind direction.

c. The measurement sequence of each location for the initial population center monitoring program is summarized in Step 6.

5. Measurement Sequence at On-site Locations

The following measurements should be made at each on-site location. The data should be entered on Form 69-9259, "Emergency Environment Monitoring Field Data Sheet".

a. Determine external γ dose rate and/or count rate.

1) Make dose rate and/or count rate measurement with the detector held about three feet off the ground (i.e., approximately at waist level). Can in a 360° direction. If the detector is so equipped, take the data both window (or shield) on and off.

<u>NOTE</u>: When using the HP-210 probe, take both shield up (GM window down) and shield down (GM window up) readings. These readings may be required later to account for sky shine. Be sure to note the correct reading under the correct heading on the Data Sheet-cpm [shield off (up)] or cpm [shield on (down)].

2) Identify the type of instrument (or probe) used, time survey was started and calibration due date of the instrument on Section I of the Data Sheet.

3) Using the highest valve obtained during the scan, calculate the net dose or count rate by subtracting the approximate background valves given in Table 4.

b. Collection of Air Particulate and Radioiodine Samples

If time and manpower considerations permit, collect an air sample of at least 10 ft^3 (30-50 ft^3 is preferred).

The sample should be drawn through both a particulate filter and a halogen cartridge. Air samples <u>must</u> be taken immediately if: 1) dose rate measurement $\geq$3 mR/hr, or 2) count rate measurement $\geq$9,000 cpm.

1) Equipment Required

a) RADECO Model HD-28B (120V AC powered) or H-809 (12V DC powered).

b) 2" diameter absolute particulate filter paper.

c) Coin envelope for retention of filter.

d) HI-Q or AgZ 2-1/4" diameter cartridge

e) Plastic bag for retention of cartridge.

f) Gummed or other label for labeling cartridge container.

g) Wristwatch or stopwatch.

2) Procedure

a) Assemble the filter and halogen cartridge in the sampling head as shown in Figure 3. Draw an arrow on the cartridge to indicate the direction of flow.

b) Place the filter head on the sampler.

c) When using an HD-28 , plug in the sampler, turn on the power, and simultaneously start a stopwatch. Reset the timer on the sampler if a sampling time of several hours is contemplated.

When using an H-809C, attach the red cable to the positive car battery terminal and the black cable to the negative and, close the hood and place the sampler on top of H to avoid engine fan turbulence. Turn the vehicle engine on. Start the sampler.

When using an H-809B, put the toggle switch on "EXT". Then connect the battery cables in the same manner as the H-809C. If an automobile is not available, use the internal battery by turning the toggle switch to "INT".

<u>NOTE</u>: The H-809B will not run using 120 VAC, so do not sample while charging.

d) Quickly adjust the flow rate to the desired value (typically 2 cfm). If the H-809 is used, do <u>not</u> attempt to adjust airflow by turning the setscrew on top of the flowmeter. Instead, note the airflow indicated on the flowmeter.

e) For a flow rate of 2 cfm, the sampling time should be at least 5 minutes.

f) Periodically check the flow indicators to verify that the flow rate is being maintained. If the flow rate changes significantly during sample collection, note the value at the end of the sample period and determine the average value of the flow rate. Record the value of the flow rate in cfm.

g) Allow the sampler to run until at least 10 ft^3 (preferably 30-50 ft^3) is collected. The greater the volume sampled, the better. Record the sample time.

h) Stop the sampler and remove the filter head.

c. Determination of gross particulate (field technique)

1) Label a particulate air sample coin envelope with the following information.

a) Location of sample

b) Date and time of start and end of sample collection. The time should be expressed using the military (24 hour period) standard.

c) Sample flow rate

d) Name of person who collected sample

2) Using tweezers, remove the particulate filter from the sample head.

3) Place the GM probe within one-half inch of the upstream side of the filter. When using an HP-210 or HP-260 probe, the reading is taken with the detector faced down. Obtain a count rate. Place the filter into the coin envelope and, using the same technique, recount the sample. If the count rate now decreases, then radon products should be suspected.

4) Use the standard background shown in Table 4 to obtain the net count rate. Calculate the gross particulate activity from the expression:

$$\mu Ci/ml = \frac{(1.59 \times 10^{-11})\ (CR_{net})}{(E_3)\ (E_f)\ (V)}$$

where:

CR_{net} = net cpm on the filter. Use the more conservative count rate if radon products are not suspected.

E_3 = probe efficiency from Table 6

E_f = filter collection efficiency, assumed to be 0.90

V = volume of sample (ft^3)

Record all data and results on the Field Data Sheet Form 69-9259.

d. Determination of gross iodine (field techniques)

1) Label a plastic bag with the following information

a) Location of sample

b) Date and time of start and end of sample collection. The time should be expressed using the military (24 hour period) standard.

c) Sampler flow rate

d) Name of person who collected sample

2) Remove the cartridge and assembly from the sample head and insert it into the adapter sample head as shown in Figure 4.

3) Open the main valve on the air cylinder. Note the pressure reading on the cylinder gauge next to the main valve. It should normally read about 1800 psi. Do not use the bottle if there is 300 psi or less. Shut the main valve. Adjust the flow control until delivery pressure gauge is set at 5 psi. Attach the air cylinder and regulator to the adapter sample head as shown in Figure 4. Open the main valve and the flow control valve. This will gently blow air through the head in the reverse direction. Continue until cylinder gauge pressure drops 200 psi. This technique removes noble gases from the cartridge.

4) Remove the cartridge from the adapter and place the probe within one-half inch of the upstream side of the cartridge. When using an HP-210 or HP-260 probe, the reading is taken with the detector faced down. Obtain a count rate and record it on the Field Data Sheet. Use the standard background shown in Table 4 to obtain the net count rate. Place the cartridge in the plastic bag.

5) Calculate the gross iodine concentration using the following expression.

$$\mu Ci/ml = \frac{(1.59 \times 10^{-11})\ (CR_{net})}{(E_2)\ (E_c)\ (V)}$$

where:

CR_{net} = net cpm on cartridge

E_2 = probe efficiency from Table 5

E_c = cartridge collection efficiency, assumed to be 0.80

V = sample volume (ft^3)

Record the concentration on Form 69-9259. It is preferable to return the cartridge to the Count Room or TSC lab if it can be done expeditiously. Otherwise use the calculation.

e. Place the coin envelope in the plastic bag with the iodine cartridge. Survey the exterior of the bag for loose contamination. Decontaminate or place another bag over it if unconditional release is not possible. Use care to prevent this bag from being contaminated. Note contamination's on label.

f. Determine personnel and equipment contamination levels prior to repacking and final movement to another area. This may entail temporary movement to low background area. If practical, remove clothing, decontaminate or bag equipment.

6. Measurement Sequence for Initial Population Center Monitoring

The following measurement program should be followed during the initial population center monitoring phase. These measurements may be preempted should the Emergency Evaluations Coordinator determine that an alternate schedule is appropriate.

a. External γ Measurements

1) Measure γ dose rate or count rate at the time that monitoring is initially established.

2) For those locations where an instrument can be left unattended, leave a Rad Owl of HPI-1010 in the integrate mode. The integrated exposure should be recorded at least every two hours.

3) For those locations where an instrument cannot be left unattended, a dose-rate or count-rate measurement should be made on an hourly basis.

b. Air Samples

1) For those locations where AC power is available and a sampler can be left unattended, place an HD-28B sampler and allow it to run continuously. Collect the first

filters and determine the iodine concentration after about 30 ft^3 have been collected. This is fifteen minutes on an HD-28B set at 2 cfm. Thereafter, collect the filters every two hours.

2) For those locations where AC power is not available, or a sampler cannot be left unattended, collect a sample every two hours of at least 30 ft^3, using a 12V dc-powered sampler operated from an automobile.

c. How to read the Pressurized Ion Chamber (PIC)

1) Equipment

(a) High Security Pin Tumbler Key (in Emergency Kit)

(b) Screwdriver with 0.25-inch blade or equivalent

2) Using the screwdriver and key (as applicable) open the door and read the liquid crystal digital display.

<u>NOTE</u>: The selector switch must be in "READ" position. Also some PIC's have a window over the display so no screwdriver is needed.

3) Look at the strip chart. One trace on it (usually a straight line) matches one of the markings immediately below the chart. This indicates the range. The second trace will normally fluctuate. This indicates the actual reading. It should correspond with that shown on the digital display. Using a felt-type pen or a pencil, indicate and write the date, time and initials of person checking the reading on the chart. If the two readings do not agree, use the higher reading.

d) Close the cover. Record data on Field Data Sheet. Report the reading.

7. Recordkeeping

All records generated by the utilization of this procedure for an exercise or emergency drill shall be forwarded the next working day to the Senior Emergency Planner for review and retention.

a. Records generated from exercises will be categorized as non-permanent and retained for a minimum of five years.

b. Records generated from actual emergency events will be categorized as lifetime and placed into lifetime storage in accordance with procedure "Requirements for Retention and Extended Storage of Operation Phase Activity Records" (AP-E-1-S1).

SUPPORTING PROCEDURES

R-2	Release of Airborne Radioactive Material
RB-11	Emergency Off-site Dose Calculations
RB-8	Off-site Emergency Radiological Environmental Monitoring
EF-4	Activation of the Mobile Environmental Monitoring Laboratory
RB-4	Access to and Establishment of Controlled Areas under Emergency Conditions
RCP-G-5	Control of Access for Radiation Protection Purposes
RCP-G-6	Release of Material's from Controlled Areas

TABLES

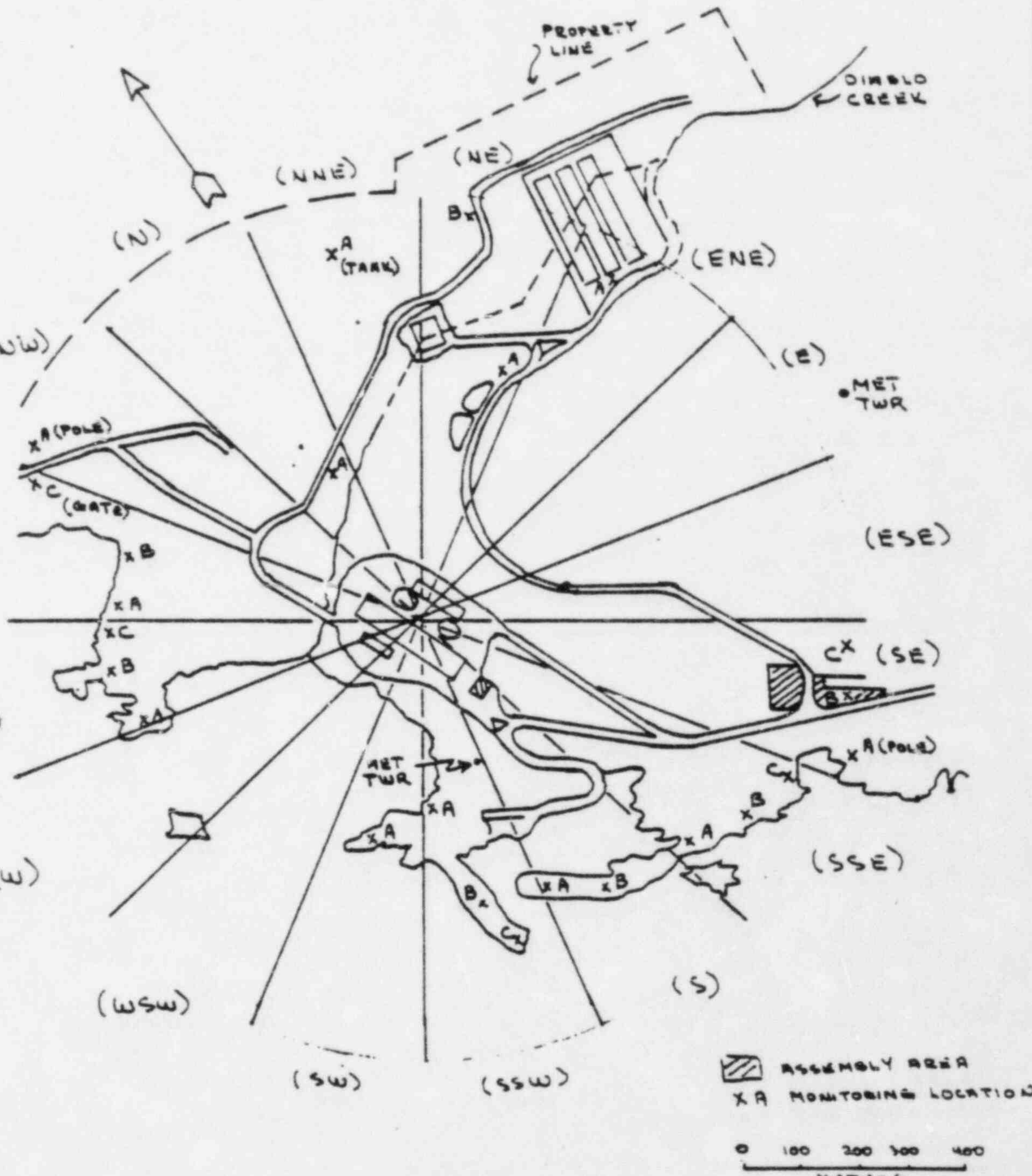
1. Description of Emergency On-site Monitoring Locations
2. Description of Emergency Off-site Monitoring Locations
3. Preferred Monitoring Locations for Initial Population Center Monitoring Program
4. Instrument Background for Beta-gamma Dose Rate Measurements and/or Count Rate Measurements (3' above ground).
5. GM Probe Efficiency Factors for Iodine Determination
6. GM Probe Efficiency Factors for Particulate Determination
7. PIC Locations

FIGURES

ATTACHMENTS

FIGURE 1

ON-SITE ASSEMBLY AND MONITORING LOCATIONS



TITLE: EMERGENCY ON-SITE RADIOLOGICAL ENVIRONMENTAL MONITORING

FIGURE 2

EMERGENCY OFF-SITE MONITORING LOCATIONS

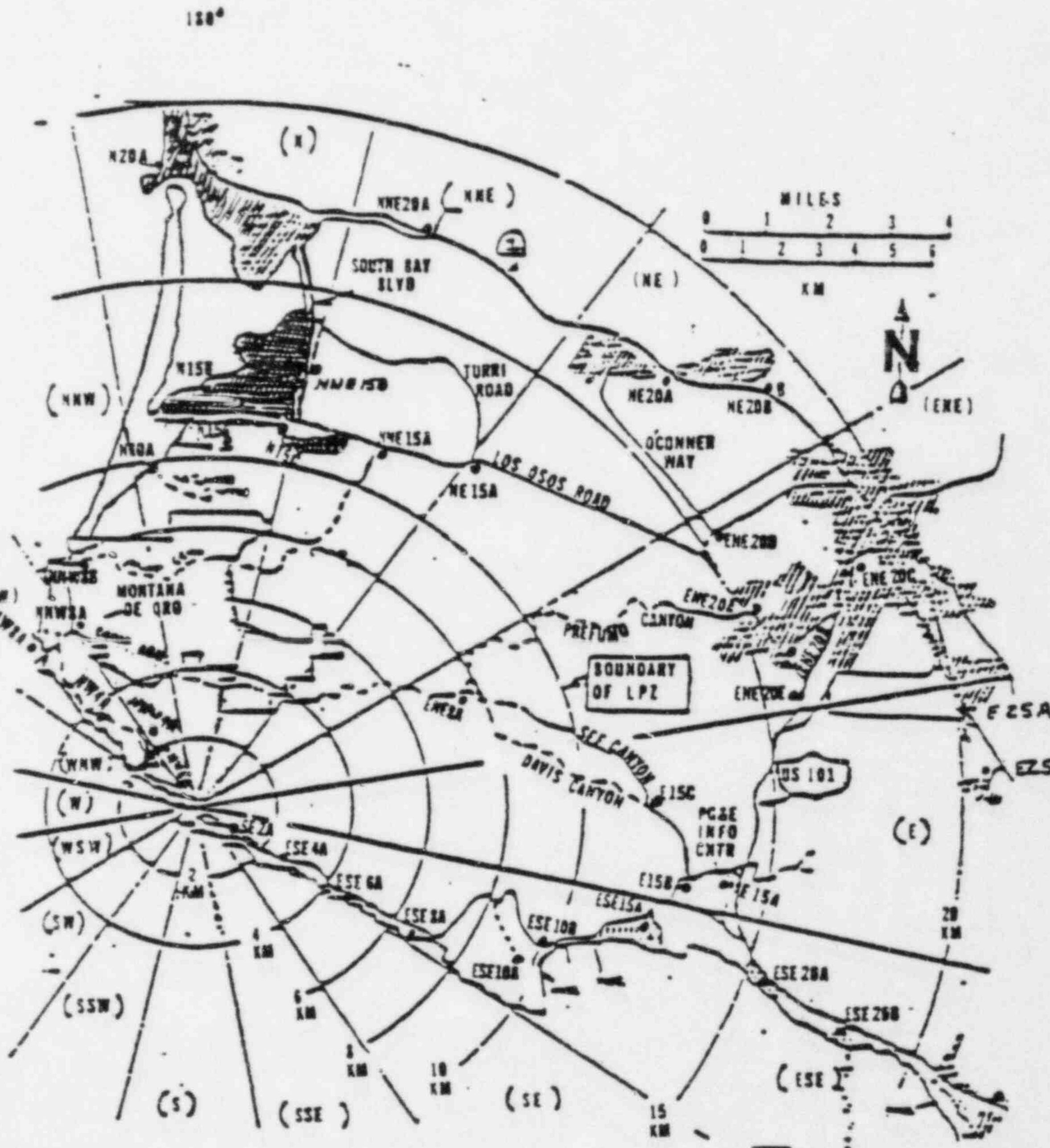


FIGURE 3

EXPLODED VIEW OF HALOGEN CARTRIDGE AND PARTICULATE FILTER IN SAMPLING HEAD

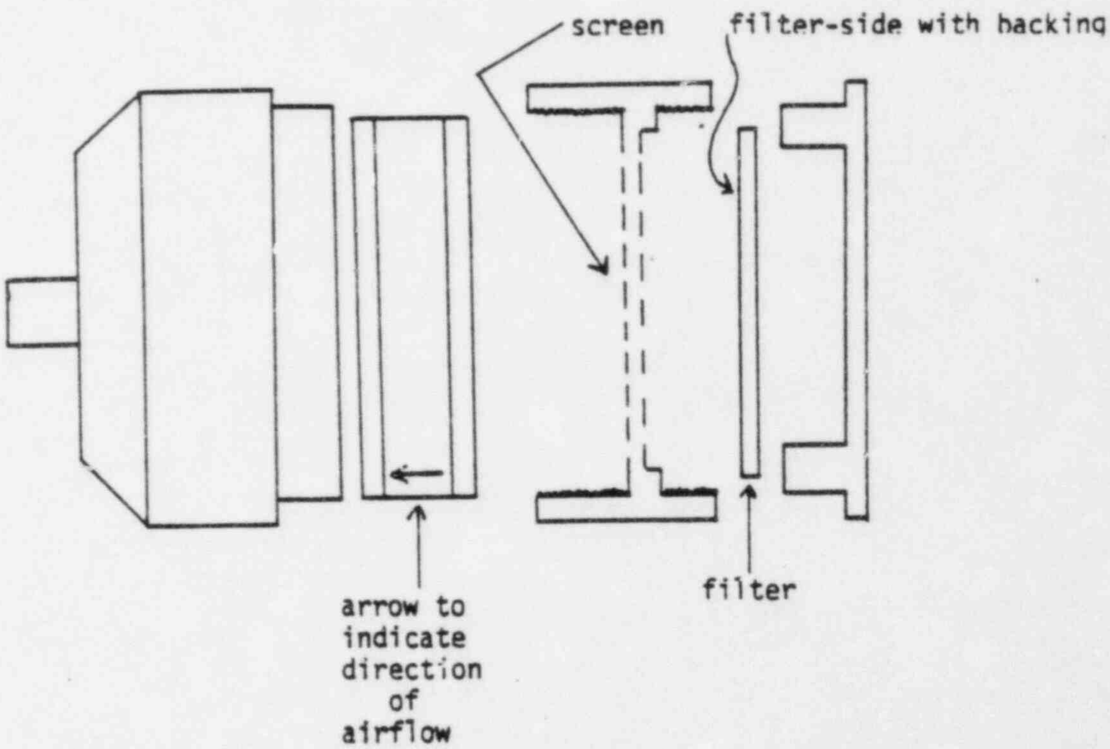


FIGURE 4

METHOD OF BLOWING NOBLE GASES FROM HALOGEN CARTRIDGE

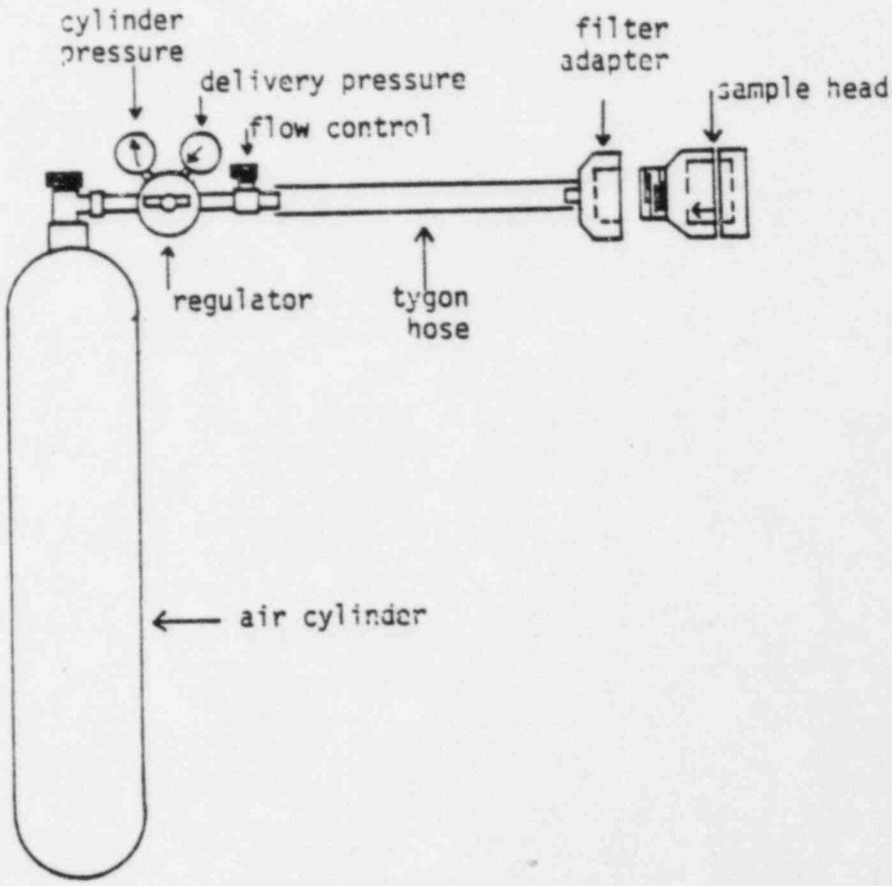


TABLE 1

DESCRIPTION OF EMERGENCY ONSITE MONITORING LOCATIONS

Coordinate	Straight Line Distance From Plant (Meters)	Description	AC Power Available
N,A	300	South side of road	No
N,NE,A	700	In front of wooden water tank	No
NE,A	420	Adjacent to east reservoir	Yes
NE,B	800	North side next to road, adjacent to gulley	No
ENE,A	700	South side of switch-yard at fence	No
SE,C	700	Air sampler-behind north side of GC warehouse	Yes
SE,B	800	Front of GC warehouse	Yes
SE,A	800	40 feet off the main road on the west side of road	No
SSE, A	700	Adjacent to guard shack on bluff	Yes
SSE,B	700	Adjacent to culvert on west side of dirt road	No
SSE,C	700	West of sandblasting area	No
S,A	---	On breakwater	No
S,B	---	On breakwater	No

TABLE 1 (Continued)

Coordinate	Straight Line Distance From Plant (Meters)	Description	AC Power Available
SSW,A	400	West of met tower - requires - 53 - key for gate - not accessible by vehicle	No
SSW,B	---	On breakwater	No
SSW,C	---	On breakwater	No
SW,A	500	West of met tower- requires - 53 - key for gate - not accessible by vehicle	Yes
WNW,A	500	Near gulley on south side - requires - 53 - key for gate	No
WNW,B	600	40 feet in front gate - requires - 53 - key	No
WNW,C	600	North side of road in clearing-may have plant growth	No
NW,A	600	North side of road in clearing-may have plant growth	No
NW,B	600	Northwest side of dirt road	No
NW,C	800	At gate on road leading to north road	No
NNW,A	800	30 feet north of steps leading to pond	No

TABLE 2

DESCRIPTION OF EMERGENCY OFF-SITE MONITORING LOCATIONS

Coordinate	Straight Line Distance From Plant (KM)	Description[8]	AC Power Available
N,10,A	9.8	West of Montana de Oro State Park sign (0.6 miles SW of Rodman Drive on Pecho Road. Good radio path off road south of State Park sign. Sign on path reads "No Vehicles".	No
N,15,A	10.0	End of Alamo Drive in Cabrillo Estates. (Turn off Pecho Road at Rodman and go to the top of the hill. Turn right onto Alamo and follow it to the end). Good radio. Phone available.	
N,15,B	11.6	Sunset Terrace Golf Course at Clubhouse. Good radio at west end of Howard Drive or on road running north of the golf course.	Yes[2]
N,15,C	11.3	Baywood Park Fire Station (Turn south off of Los Osos Road onto Bayview Heights Drive. Has a TASC-4. DER's monitoring Station 10 is located at Sunnyside School next door). Good radio. Phone available.	Yes
N,20,A	18.0	Morro Bay Power Plant. DER's monitoring Station 9 is located here. Good radio. Phone available.	Yes

TABLE 2 (Continued)

Coordinate	Straight Line Distance From Plant (KM)	Description	AC Power Available
NNE,15,A	11.3	Intersection of Los Osos Valley Road and Clark Valley Road (under PG&E transmission lines). Good radio.	No
NNE,15,B	12.9	Los Osos Jr. High School on South Bay Boulevard in the playing field. Good radio.	No
NNE,20,A	17.6	0.2 miles north along San Bernardo Creek Road is on the northeast side of Highway 1. Good radio.	No
NE,15,A	10.6	Intersection of Los Osos Valley Road and Turri Road. DER's monitoring Station 11 is located nearby. Good radio at intersection.	No
NE,20,A	17.4	Sheriff's headquarters. (EOF) Turn south on Highway 1 at sign indicating Sheriff's Operational Center. Good Radio. Phone available.	Yes
NE,20,B	19.2	PG&E substation near Men's Colony, adjacent to northeast side of Highway 1. Good Radio. Phone available.	Yes

TABLE 2 (Continued)

Coordinate	Straight Line Distance From Plant (KM)	Description	AC Power Available
ENE,8,A	9.0	See Canyon Road, 4.2 miles up from San Luis Bay Drive intersection. Good radio. Telephone available. Rattlesnake hazard.	Yes[1]
ENE,20,A	14.8	Laguna Jr. High School at intersection of Los Osos Road and Perfumo Canyon Road. Good radio.	Yes[3]
ENE,20,B	16.0	Fire station at inter- of Los Osos Valley and Madonna Roads. Good radio.	Yes
ENE,20,C	18.6	PG&E Information Zone 1 substation at corner of Walker and Pacific Streets. DER's Station 12 is also located here. Good radio.	Yes
ENE,20,D	15.6	Corner of Foothill Boulevard and O'Conner Way. Good radio.	Yes[1]
ENE,20,E	15.8	Yancy's Restaurant (formerly Hob Nob) parking lot. Good radio.	Yes[2]
E,15,A	14.5	PG&E Information Center. Good radio.	Yes
E,15,B	13.4	Bellevue-Sante Fe School. Good Radio.	Yes[5]

TABLE 2 (Continued)

Coordinate	Straight Line Distance From Plant (KM)	Description	AC Power Available
E,15,C	11.3	See Canyon Road, 1.7 miles up from San Luis Bay Drive intersection. Survey at intersection of See Canyon Road and Davis Canyon Road. Good radio.	Yes[1]
E,25,A	20.2	SLO County Airport. The field on the right of the road to the parking lot. Good radio.	Yes
E,25,B	21.5	SLO Country Club. East side of parking lot in the fairway. Good radio.	Yes[2]
ESE,4,A	2.6	Turnout on access road, 1.6 miles from Security Building. Marked with red/white fence post. Radio near plant or near location ESE,10,A	No
ESE,6,A	4.5	Turnout on access road 2.8 miles from Security Building. Marked with red/white fence post. Radio near plant or near location ESE,10,A.	No

TABLE 2 (Continued)

Coordinate	Straight Line Distance From Plant (KM)	Description	AC Power Available
ESE,8,A	6.9	Gate next to shack at road to ruins, 4.3 miles from Security Building along access road. Marked with red/white fence post. DER station 16 is near here. Radio near plant or near location ESE,10,A	No
ESE,10,A	9.6	Top of San Luis Hill[5]. Gate at 6.2 miles from the Security Building. Some radio.	No
ESE,10,B	10.0	Port San Luis Gate. TASC-4 and DER's Station 27 are located here. Good radio on road to Pirates Cove.	Yes
ESE,15,A	11.6	Parking lot behind Avila Beach Post Office. Good radio on road to Pirates Cove.	Yes
ESE,20,A	15.3	Pismo Beach Fire Dept. on Shell Beach Road. Good radio.	Yes
ESE,20,B	19.2	0.5 miles northwest of the Shorecliff Inn on Shell Beach Road. Good radio.	Yes[1]
SE,2,A	1.3	Turnout on access road, 0.8 miles from Security Building near meteorological Tower A. Marked with red/white fence post. DER Station 7 is near here. Good radio.	No

TABLE 2 (Continued)

Coordinate	Straight Line Distance From Plant (KM)	Description	AC Power Available
NW,2,A	1.6	0.6 miles north from Fieid's property gate (1 mile north from plant, just ENE of Lion Rock).	No
NW,4,A	3.5	Fields' road near large watering pond.	No
NW,8,A	6.1	Near residence by park gate.	Yes[1]
NNW,4,A	2.7	Near wood paneled house.	Yes[1]
NNW,8,A	5.8	Parking lot near end of road at southern park boundary (near gate to Fields' property). Good radio.	No[6]
NNW,8,B	7.6	Ranger station over-looking Spooner's Cove. Good radio on road south of Ranger Station at "Locked Gate Ahead" sign (near parking overlook of Spooner's Cove).	

NOTES:

[1]Power is available at nearby residences.

[2]During working hours. Also power is available at nearby residences.

[3]During school hours. Also at nearby residences.

[4]During school hours.

TABLE 2 (Notes Continued)

[5]A dirt road leads to the top of the hill. The intersection of this road with access road is marked with a red/white fence post. A 90909 key is required on the gate. A four-wheel drive vehicle is preferred. Alternatively; take reading on the access road at the marked fence post.

[6]Power is available at nearby residence on Fields' property (NW,8,A) and monitoring can be performed at this latter location is practical.

[7]During daylight hours.

[8]Radio comments refer to unaided handi-talkies use of Davis Peak Repeater (H.P. Frequency) only - other transmitter sites not considered.

TABLE 3

PREFERRED MONITORING LOCATIONS FOR INITIAL POPULATION CENTER MONITORING PROGRAM

WIND DIRECTION	DOWNWIND SECTOR	LOCATIONS[1]
CALM	---	NONE
348°45' - 11°15'	S	NONE
11°15' - 33°45'	SSW	NONE
33°45' - 56°15'	SW	NONE
56°15' - 78°45'	WSW	NONE
78°45' - 101°15'	W	NONE
101°25' - 123°45'	NNW	(NNW,8,A)
123°45' - 146°15'	NW	(NNW,8,B)
146°15' - 168°45	NNW	(NNW,8,B) (N,15,B)
168°45' - 191°15'	N	(NNW,8B) (N,15,B)
191°15' - 213°45'	NNE[2]	(N,15,D) (NNE,15,A)
213°45' - 236°15'	NE[2]	(NE,15,A) (ENE,8,A) (ENE,20,C)
236°15' - 258°45'	ENE[2]	(ENE,8,A) (ENE,20,A) (ENE,20,C)
258°45' - 281°15'	E[2]	(ENE,8,A) (E,15,B) (ESE,15,A)
281°15' - 303°45'	ESE	(E,15,B) (E,15,A) (ESE,20,B)

TABLE 3 (CONTINUED)

PREFERRED MONITORING LOCATIONS FOR INITIAL POPULATION CENTER MONITORING PROGRAM

WIND DIRECTION	DOWNWIND SECTOR	LOCATIONS[1]
303°45' - 326°15'	SE	(ESE,8,A) (ESE,15,A) (ESE,20,B)
326°15' - 348°45'	SSE	(ESE,15,A)

NOTES:

[1]See Table and Figure 2.

[2]This wind direction rarely persists for more than a few hours, so anticipate wind change.

TABLE 4

INSTRUMENT BACKGROUND FOR GAMMA-BETA DOSE RATE MEASUREMENTS AND/OR COUNT RATE MEASUREMENTS (3' ABOVE GROUND)

INSTRUMENT	BACKGROUND DOSE RATE (mR/hr)	
	WINDOW CLOSED	WINDOW OPEN
Rad Owl	0 0	
Victoreen Radgun	0.02	0.02
HPI-1010	0.015	NA
RO-2	0 0	

GM PROBE	BACKGROUND COUNT RATE (cpm)	
	SHIELD ON	SHIELD OFF
HP-240	60	60
HP-260	NA	50
HP-210	NA	50

TABLE 5

GM PROBE EFFICIENCY FACTORS, ε_2 FOR IODINE DETERMINATIONS

GM PROBE	ε_2 (counts/dis)
HP-210	0.09
HP-240/270	0.013
HP-260	0.09

TABLE 6

GM PROBE EFFICIENCY FACTORS, ε_1, FOR PARTICULATE DETERMINATIONS

GM PROBE	ε_3 (counts/dis)
HP-210	0.18
HP-240/270	0.018
HP-260	0.20

TABLE 7

PIC LOCATIONS

LOCATION	DESCRIPTION
Site 1	DCPP North Gate Guard Post
Site 2	SSW Corner of Target Range
Site 3	715 Harbor Street, Morro Bay. Small fenced yard on NE corner of Harbor Street and Piney Street intersection.
Site 4	Montano de Oro State Park. At the Park Ranger's residence, adjacent to the emergency siren.
Site 5	Los Osos Fire Department
Site 6	SLO County Sheriff's Office. South of EOF trailer behind retaining wall
Site 7	SLO Police Department. Intersection of Santa Rosa Street and Walnut Street. Behind fence SW of Walnut Street driveway.
Site 8	SLO Service Center
Site 9	PGandE Energy Information Center. Take San Luis Bay Drive exit from U.S. 101. On hill above employee parking lot.
Site 10	DCPP Front Gate
Site 11	Pismo Beach. From Bellow Street go N.E. onto Main Street. Turn right at first road to the water storage tank. On top of hill N of pump house and W of water storage tank.

TABLE 7 (Continued)

LOCATION	DESCRIPTION
Site 12	SLO County Building. Grover City. SW corner of SLO County Social Services Building on Longbranch Street, Grover City.

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

EMERGENCY ENVIRONMENTAL MONITORING FIELD DATA SHEET

Team ______ Leader ______ Member ______

Monitoring Location ______

1. THREE FOOT BETA-GAMMA RADIATION FIELD READINGS

a. Count Rate

Calibration Due Date ______

Time	Type of Probe	CPM[Shield off(UP)] Gross	CPM[Shield off(UP)] BKG*	CPM[Shield off(UP)] Net	CPM[Shield on(down)] Gross	CPM[Shield on(down)] BKG*	CPM[Shield on(down)] Net

b. Dose Rate

Calibration Due Date ______

Time	Instrument	mR/hr(Window Open) Gross	mR/hr(Window Open) BKG*	mR/hr(Window Open) Net	mR/hr(Window Closed) Gross	mR/hr(Window Closed) BKG*	mR/hr(Window Closed) Net

c. Integral Dose

Instrument	① Time Started	② Time Complete	③ Duration(HR) ② - ①	④ Total Dose(mR)	⑤ Dose Rate(mR/hr) ④ ÷ ③

2. AIR SAMPLE DATA

Calibration Due ______

Sampler	Time Started	Time Completed	Duration (Minutes)	Flow Rate (CFM)	Sample Volume (FT^3)

3. PARTICULATE DETERMINATION

Type of Probe	Gross	CPM(Shield off) BKG*	① Net	② ϵ_i	③ E_f	④ Volume(FT^3)	$\frac{① \times 1.59 \times 10^{-11}}{② \times ③ \times ④}$ ($\mu Ci/ml$)

4. IODINE DETERMINATION

Type of Probe	Gross	CPM(Shield Off) BKG*	① Net	② ϵ_i	③ E_c	④ Volume(FT^3)	$\frac{① \times 1.59 \times 10^{-11}}{② \times ③ \times ④}$ ($\mu Ci/ml$)

*NUMERICAL VALUE FOUND IN EMERGENCY PROCEDURE RB-7 OR RB-6

5. ENVIRONMENTAL MONITOR READINGS

a. PIC Reading

Calibration Due ______

Time	Dose Rate(mR/hr)

b. TASC-4 Reading Calibration Due ______

① Scaler Count	② Count Time (Sec)	① ÷ ② (mR/hr)

6. GROUND SURVEYS

Time ______

Description	Probe	CPM(Shield off) Gross	BKG*	Net ①	② ϵ_1	① x ② (µCi/ml)

7. VEGETATION SAMPLES

NOTE: USE HP-240 OR EQUIVALENT PROBE

Time ______

Description	CPM(Shield off) Gross	BKG*	Net ①	① x 2.5×10^{-5} (µCi/ml)

8. SMEAR SAMPLES

Time ______

Description	Probe	Gross	BKG*	Net ①	② ϵ_3	3 Area Smeared (Ft²)	$\frac{0.11 \times ①}{② \times ③}$ (µCi/dm²)

9. LIQUID SAMPLES

Time ______

Description	Volume of Sample Counted	Immersion Data Gross CPM	BKG*	Net CPM

10. REMARKS

TITLE EMERGENCY PROCEDURE
CALCULATION OF RELEASE RATE
AND INTEGRATED RELEASE

IMPORTANT TO SAFETY

APPROVED [signature] 2-2-84
PLANT MANAGER DATE

SCOPE

This procedure describes methods to determine airborne radioactivity release rates and total curie release under emergency conditions. The methods to determine airborne radioactivity release rate and total release are discussed as follows: 1) Determination of total release and release rate via the plant vent; 2) Determination of total release and release rate via the plant containment dome monitors; 3) Determination of total release and release rate via the secondary system (i.e., steam dump, blowdown tank) or an unmonitored pathway. This procedure and changes thereto requires PSRC review.

GENERAL

Figure 1 provides a flow chart to determine which of the above methods is appropriate for use under existing emergency conditions. Work sheets to perform necessary calculations are provided in this procedure. Release rate and total curie release values will be used to calculate projected doses offsite (EP RB-11).

PROCEDURE

Work Sheets 69-9260, 69-9284, 69-10555, and 69-11105 of this procedure are to assist in making the calculations. The instructions which follow are in the same sequence as they appear on the work sheets.

1. Determination of Noble Gas Release Rate Via the Plant Vent RE-14 or RE-29

   a. Determine the Plant Vent Flow Rate

      This section presents the method(s) to be utilized in determining the plant vent flow rate. The information obtained should be recorded on the work sheets as described below:

1) The plant vent flow rate can be read directly on FR-12 (located on the Unit 2 RMS board in the control room). Enter the flow rate (in cfm) at Step 2.B on the work sheet.

2) If FR-12 is inoperable, the flow rate may be estimated by multiplying the number of fans in operation on the affected unit by their respective flow rates (Step 2.A on work sheet) and summing.

3) Subsequent calculations require the flow rate to be expressed in cc/sec instead of cfm. This conversion is carried out in Step 2.C.

b. Determine the Plant Vent Noble Gas Release Rate

The release rate will probably exhibit considerable variation over the course of the release. However, at any moment in time, the release rate can be determined as follows:

1) Determine which noble gas plant vent monitor is on-scale. RE-14 (normal range) reads in cpm. RE-29 (high range) reads in mR/hr. Enter the reading on Work Sheet 69-9260, Step 3.

2) Using Figure 2 or 3, as appropriate, convert the monitor reading to plant vent concentration (μCi/cc). Enter the value on Work Sheet 69-9260, Step 3.

<u>NOTE</u>: On Figure 2, use the curve labeled "effective age < 1,000 hours" for all accidents except those involving spent fuel which has been stored for greater than 1,000 hours ($\simeq$ 1-1/2 months).

3) Release rate may be obtained by using the following formula:

$$\dot{Q}\ (\text{Ci/sec}) = \text{Vent Flow Rate (cc/sec)} \times \text{Vent Concentration } (\mu\text{Ci/cc}) \times 10^{-6}\ (\text{Ci}/\mu\text{Ci}) \quad (1)$$

Enter result on Work Sheet 69-9260, Step 3.

2. Determination of Total Curie Release of Noble Gases Via the Plant Vent (RE-14 or 29).

   Total curie release of noble gases is the product of release rate times the duration of release (ΔT).

   a. From the strip chart of the monitor of interest, determine the duration of the release rate (ΔT) in seconds. Enter value on Work Sheet 69-9260, Step 4.A.

   b. Multiply Δt time $\dot{Q}$ (Ci/sec) to obtain total release. Enter value on Work Sheet 69-9260, Step 4.

   c. If the release rate has varied over time, sum the products of Δt times $\dot{Q}$ to obtain the total curie release up to that point in time. Enter value on Work Sheet 69-9260, Step 4.

3. Determination of the Total I-131 Release and Release Rate Via Plant Vent (RE-24).

   Each Unit's plant vent is sampled by a normal range radioiodine monitor (RE-24). This instrument measures both the I-131 collection rate and the I-131 activity collected on the silver zeolite cartridge. Either of the above values can be displayed locally by operating a selector switch on the instrument. It is important to note that this instrument only measures the I-131 activity collected on the cartridge. To convert cartridge collection figures to total plant release, it is necessary to multiply by the ratio of the plant vent flow rate to the sample flow rate.

   To use this instrument for assessing an accidental release, proceed as follows:

   a. Estimating Curies Released and the Release Rate to the Present Moment

      1) If the high alarm for RE-24 has activated, go to Step 3.C.

      2) Read the sampler flow rate on the local indicator. This data is entered in Section 5.A of the Work Sheet

and is used to calculate the vent/sampler flow ratio. The plant vent flow rate (use the cfm value from Section 2.B of the Work sheet) is also entered and the ratio is determined.

3) Turn the toggle switch on the front of the instrument to the "$\mu Ci(x10^{-6})$" position.

4) Turn the SCALE FACTOR switch to the "X1000" position. This switch is located inside the door on the lower part of the front of the instrument. Enter the scale factor in Section 5.B of the Work Sheet.

5) Wait at least one minute (to allow for the instrument response time) and then read the chart and enter this reading on the work sheet. The chart reading will be a number between 1 and 1,000. Do not apply any other correction factors to the chart reading, since these are all taken into account on the work sheet.

<u>NOTE 1</u>: If the instrument does not read upscale with a "X1000" scale factor, I-131 will not be a significant contribution to exposure projections from the accident, and calculations of I-131 dose rates are performed primarily for informational/documentational purposes. Do not make any more adjustments to the instrument until the engineering staff arrives.

<u>NOTE 2</u>: If the release produces an off-scale reading on the "X1000" scale, I-131 will have to be assessed using the high range iodine monitor; or alternatively, the cartridge will have to be analyzed in the counting room.

6) The activity release to the present time is the product of the chart reading, the scale factor, the flow rate ratio, a conversion factor of 10^{-12}, and an iodine plateout factor of 1.1*. The latter factor includes

*(Transmission=1-iodine plateout). Based on a 0.3 micron particle size the plateout is 7.7%. The resulting transmission is 0.9, which results in a correction factor of 1.1 for the purposes of emergency release rate calculation.

the 10^{-6} factor which is to be applied to the chart reading when the switch is in the "$\mu Ci(x10^{-6})$" position, and a second factor of 10^{-6} to convert µCi into Ci.

7) The I-131 release rate (Ci/sec) is the product of the chart reading, the scale factor, the flow rate, a conversion factor of 10^{-17}, and an iodine plateout factor of 1.1*.

b. Projecting the Total I-131 Release

Once the activity released to the moment is known, there are three ways to project the total I-131 release.

1) Estimate the total release duration after assessing the situation.

2) Extrapolate the total release duration from the trace on the chart.

<u>NOTE</u>: This may not be possible if the scale factor was changed during the release.

3) Calculate the total release based on the "cleanup half-life."

a) Move the toggle switch to the "$\mu Ci/sec(x10^{-11})$" position. <u>Wait 1 minute</u> and read the chart, then return the switch to the "$\mu Ci(x10^{-6})$" position.

b) The additional curies that will be released are obtained by using the following equation:

$$Q_{ADD} = (1.44)\ \dot{Q}_{131} \times t_{1/2,CU} \qquad (2)$$

where:

Q_{ADD} - additional curies (Ci) that will be released

$\dot{Q}_{131}$ = I-131 release rate (Ci/sec)

$T_{1/2,CU}$ = cleanup half-life (seconds)

*See Page 4

c) Determining the Cleanup Half-Life.

Given any two release rates during the cleanup period (decreasing release rate) and the time interval between them, the cleanup half-life is given by:

$$T_{1/2,CU} = \frac{(0.693)\ \Delta t_{2,1}}{\ln (\dot{Q}_{2,CU}/\dot{Q}_{1,CU})} \qquad (3)$$

where:

$T_{1/2,CU}$ = cleanup half-life (seconds)

$\Delta t_{2,1}$ = elapsed time between data points 1 and 2 (seconds)

$\dot{Q}_{1,CU}$; $\dot{Q}_{2,CU}$ = release rates at times 1 and 2 respectively (Ci/sec)

If for any reason the cleanup half-life cannot be determined, assume the following values.

Location of Release	Cleanup Half-Life in Seconds as a Function of Number of Exhaust Fans in Operation		
	1	2	6
Auxiliary Building	600	300	-
Fuel Handling Bldg.	560	280	-
Containment	1980	-	-
Turbine Building	7560	3780	1260

d) The total release is obtained by adding the curies already released to the projected curie release.

$$Q_{TOTAL} = Q_{TO\ MOMENT} + Q_{ADD}$$

c. Determination of Total I-131 Release if RE-24 High Alarm Activates

1) Send a qualified person equipped with a portable dose rate instrument to the RE-24 local readout and recorder.

2) Use portable dose rate instrument to determine general background levels of radiation in the vicinity of the monitor. Record readings and their locations on Work Sheet, Section 6.A.

3) Turn off sampler flow. Record total time flow is off, on Work Sheet, Section 6.B.

4) If radiation levels are high, do not open cartridge door. Record dose rate outside the shield at distance of measurement on Work Sheet 6.C.

   a) Estimate distance from center of cartridge to point where dose rate is measured. Record value (in meters) on Work Sheet 69-9260, Step 6.C.

   b) The equation for the shield closed method is based on a source term involving a mixture of radioiodines representing unfractionated core inventory following two years of full power operation.

   $$Q = 60\ DR(d)^2(IP) \qquad (4)$$

   where,

   Q = quantity of radioiodines collected on cartridge (Curies)

   DR = measured dose rate at distance of interest minus general background dose rate (R/hr)

   d = estimated distance from center of cartridge to point where dose rate is measured (meters)
   IP = Iodine plateout in sample lines = 1.1*

5) If radiation levels permit, open the cartridge door to make dose rate measurement. Record value on Work Sheet, Section 6.D.

   a) Estimate distance from center of cartridge to point where dose rate is measured. Record value (in meters) on Work Sheet, Section 6.D.

*See Page 4

b) The equation for the shield open method is:

$$Q = 1.3\ DR(d)^2(IP) \qquad (6)$$

where,

Q = Quantity of radioiodines collected on cartridge (Curies)

DR = dose rate measured at distance of interest minus general background dose rate (R/hr)

d = estimated distance from center of cartridge to point where dose rate is measured (meters)

IP = Iodine plateout factor = 1.1*

6) Replace cartridge and turn on sampler flow.

7) Transport cartridge to laboratory for radio-analysis.

d. Determination of I-131 Release Rate if RE-24 Alarm Activates.

1) Release rate of radioiodines in the plant vent effluent can be calculated from the following equation:

$$\dot{Q} = (Q/\Delta t)(F/F_s)(IP) \qquad (7)$$

where,

$\dot{Q}$ = radioiodine release rate in plant vent effluent (Ci/sec)

Q = Quantity of radioiodines collected on cartridge (Curies)

Δt = time over which release rate is averaged (seconds)

F = plant vent flow rate (CFM)

Fs = sampler flow rate (CFM). Normally equal to 1.0 CFM.

IP = Iodine plateout factor = 1.1*

*See Page 4

2) The time over which the release rate is averaged (Δt) should be taken as the time during which the cartridge was installed in the monitor and sampler flow turned on and RE-24 high alarm activated. Some judgment must be used to properly estimate this time period.

4. Determination of Release Rate and Total Curie Release from Containment Radiation Monitor Readings (RE-30 and 31).

Form 69-10555 has been provided to assist in making these calculations.

a. Determine Noble Gas Release Rate

1) Containment High Range Area Monitor Readings

a) Determine the time in hours which has elapsed since the accident onset. Record this on the work sheet, in Sections 1.A through 1.D, and 2.C through 2.D.

b) Record the readings from both high range monitors (RE-30 and RE-31) on the Work Sheet, Section 1.A.

(1) If both monitors are functioning correctly, and neither is suspected of being near a "hot spot", then add the readings together, and record the sum on Work Sheet 69-10555, Section 1.A. Divide the sum by 2.0 and record the sum on the Work Sheet, Section 1.A, as "Mean Exposure Rate."

(2) If one monitor is not working properly, or is suspected of being near a "hot spot", record the properly operating monitor's reading in the "Mean Exposure Rate" column of Section 1.A.

2) Using the elapsed times since the accident began recorded on the work sheet, determine the Design Base Accident (DBA) - Loss of Coolant Accident (LOCA) exposure rate (R/hr) from Figure 4. Record this value in the "Actual Exposure Rate" column of Section 1.B of the Work Sheet.

3) Determine the corresponding noble gas release rate (Ci/sec) for the LOCA model at time t(hr) from Figure 5. Record this value in the "DBA-LOCA Noble Gas Release Rate" column of Section 1.C of the Work Sheet.

4) Divide the actual containment mean exposure rate by the DBA-LOCA exposure rate. Multiply the result by the DBA-LOCA Noble Gas release rate. The resultant value is the estimated noble gas release rate. Enter this value in Section 1.D of the Work Sheet.

b. Determine the Total Curie Release of Noble Gases

c. Determine the I-131 Release rates

1) Containment High Range Area Monitor Readings

Use the values for "Actual Containment Mean Exposure Rate" method determined in Work Sheet 69-10555, Section 1.A.

2) Design Basis Accident Exposure Rates and Release Rates

a) Determine the "DBA-LOCA exposure rate" values from Figure 4 (or use the values determined above).

b) Determine the corresponding "DBA-LOCA I-131 Release Rate" (Ci/sec) from Figure 6 and enter on the Work Sheet, Section 2.C.

3) Calculation of Accident I-131 Release Rates

a) Enter the values of "Ration of Actual/DBA Exposure Rates" and "DBA-LOCA I-131 Release Rate" for time (t) in Section 2.D of the Work Sheet.

b) Multiply the "Ratio of Actual/DBA Exposure Rates" by the DBA-LOCA I-131 Release RAte. The result is the "Projected I-131 Release Rate for time = t." Enter this value in Section 2.D of the Work Sheet.

5. Estimation of the Curies Release Via Secondary System or Unmonitored Pathway

A work sheet for determining the radioactive releases that may be associated with a steam generator tube rupture is provided as For 69-11105, "Worksheet for Use of Main Steam Line Monitors, Or RCS Coolant Sample Results During S/G Tube Rupture Accident." The worksheet provides instructions for the use of monitors RE-71, RE-72, RE-73, and/or RE-74 to determine releases that occur via the steam-line safety valves. Alternative methods to the use of RE-71, 72, 73, and 74 are also provided. If the condenser is still in service during a steam generator tube rupture accident, then release rate may be determined using the plant vent monitors described in this procedure. In the event that all these methods of assessing radiological release rate fail, or if it is determined that an unmonitored release path from the plant exists, then follow the instructions provided in this section to estimate the release rate.

To assist in making these calculations, Form 69-9284, "Worksheet for Estimation of Curie Release" has been developed. The worksheet is laid out in the same sequence as the discussion that follows.

a. Determination of Available Activity from a Fuel Release

This section of the procedure should be used in the event of a fuel handling accident, LOCA, or other accident where the major portion of the release comes from fuel damage.

1) Base Data

a) For each assembly involved (or the whole core if it was involved) enter the last date at which power operation occurred.

b) The cooling time is the interval (days) between the time of last power operation and the time that the release occurred. Cooling time is utilized to correct the fission product inventory in the damaged core for decay. For a bundle which has been out of the core 3 months, t_{cool} = 90 days. For a LOCA during operation, t_{cool} = 0 days.

c) The number of damaged fuel rods (suffering either clad rupture or pellet melting) must be estimated. For reference purposes, each assembly consists of 264 fuel rods, and the core as a whole contains 50,952 fuel rods (193 assemblies).

d) Enter the average reactor (not bundle) power over the last 30 days of reactor operation. For example, if a bundle is damaged and, during its last 30 days of operation, the reactor operated at 100% power 75% of the time and 50% power 25% of the time, the average power (expressed as a fraction of rated power) is:

$$(1.00)(0.75) + (0.50)(.25) = 0.875$$

2) Available Activity from Gap Release

If there is no pellet melting, the activity which is released is assumed to be gap activity only.

a) Section A.2., column 1 Form 69-9284 gives the equilibrium gap activity for the peak flux region of the core from full power operation. Since the highest assembly may not be involved, use of this data will usually overestimate the gap activity.

b) The full power values (column 1) are multiplied by the fractional core power, determined previously, to estimate the fraction of equilibrium gap activities. Since Kr-85 has a very long half-life (10 years), it builds up slowly through the cycle and is relatively insensitive to short term fluctuations in the reactor power. Therefore, for conservatism, the full power end of the cycle value is used.

c) In order to evaluate the radioactive decay which occurs during the interval between reactor shutdown and the time the accident occurs, first calculate the number of half-lives which have passed by determining the ratio $t_{cool}/T_{1/2}$ (enter

result column 3). The isotope half-lives are: Xe-133 = 5.3 days, I-131 = 8 days (since Kr-85 has a 10 year half-life, this ratio is set equal to zero, and always is assumed to have a decay factor equal to 1.0). The decay factor is then obtained by use of Figure 7 (enter result column 4, Section A.2.).

d) It is assumed that 100% of the noble gases are released from the gap of any fuel rod which suffers cladding failure. Since much of the iodine plates out, it is assumed that only 25% is released. This assumption is accomplished by multiplying the total gap activity by a gap release fraction of 0.25 (enter result column 5, Section A.2.).

e) When the foregoing factors are multiplied together and multiplied by the estimated number of rods with ruptured cladding, the total number of curies released from the fuel rod gaps is obtained (enter result column 7, Section A.2.).

3) Activity from Fuel Rod Melting

Section A.3., column 1 of Form 69-9284 gives the equilibrium activity for the peak power rod at full power operation. Calculation of the available activity from fuel rod melting is the same as the calculation of gap release. The only difference is that the curies of activity available for release from the fuel far exceeds that available from the gap.

4) Total Available Activity from Fuel Damage

This calculation consists of summing the foregoing results from gap release and fuel rod melting.

b. Determination of Available Activity from a Gas Release

The second type of release mechanism results from situations where noncondensible gases constitute the principal material released. If a liquid is present in the container which

fails, conditions are assumed to be such that the liquid does not vaporize significantly (although it may give up dissolved gases due to depressurization). This type of release is characteristic of a volume control tank (VCT) rupture, or other waste tank failure.

1) Base Data

The parameters of interest here are:

a) The concentration of activity in the container evaluated at atmospheric pressure.

b) The absolute pressure of the tank in psia, required for later corrections for gas compressibility.

c) The volumes of liquid and gas in the container, where the volume of gas is evaluated at the initial tank pressure. The volumes are expressed in cc. Volume conversion factors are as follows:

$$1 \text{ gal} = 3,785 \text{ cc}$$

$$1 \text{ ft}^3 = 23,317 \text{ cc}$$

The accident summary sheets give tank volumes for tanks involved in the various analyzed accidents.

d) The decay time, the interval in days, from the time the tank contents were sampled and analyzed and the time the release occurred.

2) Calculation of Available Activity

a) The sample concentrations in µCi/cc at atmospheric pressure are entered in column 1 of Part B.2. of Form 69-9284.

b) Since it is standard practice to express gas sample results in µCi/cc at <u>atmospheric</u> pressure, the concentrations in column 1, Section B.2. must be converted to prevailing tank pressure. This is

accomplished by multiplying by the ratio, Pcon/14.7, where Pcon is the absolute container pressure in psia. No pressure corrections are required for the liquid sample concentrations.

d) The initial activity must be corrected for decay. First, the number of half-lives which have elapsed is calculated in Section B.2., column 5, and then the decay factor is obtained using Figure 7. The isotope half-lives are: Xe-133 = 5.3 days; I-131 = 8.05 days. The decay factor for Kr-85 is approximated as being equal to one (1.0) due to its long half life of 10 years.

If additional activity has been added since the last sample was taken, then decay corrections should be used with care. For example if activity has been added on an irregular basis, then an estimate of this activity addition, corrected by its decay factor, should be added to the activity obtained by the above procedure. However, if activity is continuously added, a container's activity should be assumed to be in equilibrium and equal to the activity present at the time of the last sampling, and no decay factor should be applied.

e) All of the noble gas is assumed to be released from the gas filled void. Also, any noble gases dissolved in the liquid are assumed to bubble out of the liquid and be released when the container fails. However, iodine which is dissolved in water tends to remain in the liquid phase. In a liquid which does not boil, only 1 part in 10,000 will degas from the liquid. Thus we have to multiply the quantity of iodine by a factor of 10^{-4} to calculate what is actually released.

f) The available activity in Ci, Section B.2., column 9, is obtained by multiplying the initial container activity (μCi) by the decay factor, the liquid release fraction, and a conversion factor to convert μCi to Ci.

c. Determination of Available Activity from a Steam Release

This third type of calculation is employed when the accident scenario is a steam release or a hot liquid which flashes to vapor as it is released. Examples of this type of release are: steam line break, small loss of coolant without fuel damage (if fuel is damaged, part A is used to estimate the release from the fuel), blowdown through relief valve, feedwater line break, letdown line failure, etc.

1) Base Data

the principal information which must be estimated is the mass of steam which is released. The total mass contained in various containers is given in the accident summaries of Attachment 2. However, the entire contents of a container may not be released, so the circumstances of the actual accident must be considered. One way to estimate the mass release is to utilize the blow down rate and duration, if known. Another way is to compare the "before release" and "after release" contents of the container.

If the container initially contains both liquid and steam, and the release is from a point above the water level, assume that the initial inventory of steam is exhausted before any liquid vaporizes. Then, the difference between the total mass release and the initial inventory of steam represents the mass of liquid which is vaporized. If the release comes from a point below the water level, assume all of the released mass is from vaporized liquid. If the container is completely filled with water, then all of the mass release is considered to be vaporized water. The reason for making these distinctions is that the retention of iodine is influenced by the way in which the release takes place, as will be seen later, in Section 5.C.2.c) of this procedure.

2) Calculation of Available Activity

a) The mass of steam and vaporized liquid which are released are entered in Section C.2., column 1.

The pounds of steam released refers only to the initial inventory of steam when a break occurs above the water level. Otherwise, the release is considered to be vaporized water (see previous discussion).

b) In Section C.2., column 2, enter the sample results. The sample will be either as µCi/gm or µCi/cc of condensed liquid. The two sets of units are equivalent since the latter case refers to cc of water at STP, which has a density of 1 gm/cc.

c) In column 4, enter the release fraction. Basically, 100% of the Xe and Kr contained in the blown down mass is assumed to be released, regardless of whether it is steam or water, so the release fraction is 1.0. In the case of iodine release from water, the solubility of iodine makes it tend to stay with the water. When the leak is from a point above the water level, the appropriate release fraction for iodine from water is 10^{-2}. When the leak is from a point below the water level and the water itself tries to escape, it carries more iodine with it and the appropriate factor is 10^{-1}.

d) The available activity is the product of the previous factors.

In hot water systems such as the primary and secondary systems, activity is continuously added to the system due to fuel leaks, tube leaks, etc. Thus, equilibrium activity concentrations are rapidly established in such a system, and it is not necessary, or appropriate, to take credit for a radioactive decay factor.

d. Determination of Initial Releasable Activity from Available Activity

In the previous three subsections, we determined the activity available for release from the failed container, i.e., the "available activity". All of this activity which

is available for release may, or may not, actually be released. In the case of noble gases, the only effective cleanup mechanism is radioactive decay. In the case of iodine release, however, there are several additional cleanup mechanisms.

These include filtration, water scrubbing, and containment sprays. Credit in a given situation should be taken for these cleanup methods as appropriate in calculations of noble gas and iodine releases. Some release cleanup mechanisms such as exhaust filtration and water scrubbing are independent of radioactive decay and are treated separately.

However, the building volume cleanup processes such as building ventilation exhaust and containment spray are time-dependent and must be treated simultaneously with radioactive decay.

1) Total Available Activity

Total available activity is the sum of the available activities from all applicable modes of release, (i.e., fuel pellet, cladding gap, or reactor coolant for a fuel melt accident). This activity is entered in column 2 of Part D of Form 69-9284.

2) Water Scrubbing Factor

If an iodine release occurs in water, such as from damage to a fuel assembly while submerged in the spent fuel pool, a large part of the iodine will be scrubbed ut of the bubbles as they rise to the surface. If the release is above the water level, use 1.0. If the release is below the water level use 0.01 (the value used in the design basis analysis). The most realistic value is 0.0013 (factor of 760).

3) Filter Cleanup Factor

Theoretically, a charcoal filter has a removal efficiency of 99.97%, corresponding to a cleanup factor of 0.0003. However, some of the iodine is organic, which is not removed as efficiently, so 99.9% (filter

factor of 0.001) is assumed. The design base analyses in the FSAR assume only 90% efficiency (clean-up factor of 0.1). The appropriate filter exhaust cleanup factor for iodine should be entered in column 4, Section D.

NOTE: Be certain that an exhaust filter cleanup factor is only taken credit for if the plant ventilation is operating in the affected area during the release.

4) Initial Releasable Activity

The initial releasable activity is obtained by multiplying the total available activity, column 2 of part D of Form 69-9284, by the applicable factors in columns 3 and 4.

Although this product is called the initial releasable activity, it must be remembered that if the release occurs within the containment, and containment integrity is intact, then the actual release will be significantly lower than this amount due to containment sprays and radioactive decay during the release period. For releases that occur in other buildings, the actual release to the environs will still be somewhat lower than the "releasable activity" due to radioactive decay during the release period. Therefore, it is necessary to compute an effective cleanup half-life which accounts for these removal processes.

5) Effective Cleanup Half-Life

The effective cleanup half-lives are determined in Section E of Form 69-9284 for the case of releases treated by exhaust ventilation or containment spray. If these systems are not operating, credit cannot be taken. The radioactive decay removal rates are entered in column 2 for use here. The building free volume for the appropriate activity release building is obtained by referring to Note 8 of Form 69-9284 and entered in column 3. The ventilation exhaust flow for the affected building is obtained using Work Sheet 69-9260 and entered in column 4 of Form 69-9284. The exhaust vent removal rate of activity from the building is

computed by dividing column 4 by column 3. This value is entered in column 5. In order to obtain an exhaust vent removal rate with the unit of hr $^{-1}$, the value in column 5 must be multiplied by 3,600 sec/hr. This result is entered in column 6. The containment spray removal rate is next entered in column 7 for I-131. The appropriate values are zero if no pumps are functioning, 31.5 hr $^{-1}$ if one pump is functioning, and 92.4 hr $^{-1}$ if two spray pumps are operating as indicated by Note 7 on Form 69-9284. The total building removal rates are next entered in column 8 by summing columns 2, 6, and 7. Finally, the effective cleanup half-lives are determined by diving 0.693 by the total building removal rates in column 8. The effective cleanup half-lives obtained are entered in column 9.

6) Initial Release Rate

The initial release rate at the time (t_1) of the release is obtained by first transferring the initial releasable activity at time t_1 from column 5 of Section D to column 2 of Section G of Form 69-9284. If the released activity is being vented through the station's exhaust ventilation system, then the appropriate exhaust vent removal rates from column 5 of part E are entered in column 3 of Section G. If the activity was released to the reactor containment and the containment ventilation system is isolated, then zero must be entered in column 3 for each isotope. If the containment ventilation system is isolated, then zero must be entered in column 3 for each isotope. The containment design basis leakage rate has been entered in column 4 for such cases. The building vent initial release rates are calculated by multiplying the values in column 2 by the values in column 3. The resulting values are entered in column 5. If the activity was released to an isolated containment, then the initial release rates are obtained by multiplying column 2 by column 4. These release rates are then entered in column 6.

7) Subsequent Release Rates

In order to determine the release rate at some subsequent time (t_2) it is necessary to first determine

the releasable activity at the corresponding time. Two separate cases for computing the releasable activity at subsequent time t_2 are identified and described in the following description of release rate calculations.

a) Effective cleanup half-lives do not change between t_1 and t_2

If the release rate at some subsequent time t_2 is desired and the total building removal rate (see column 8 of part E) has been constant between t_1 and t_2, then the release rate at time t_2 can be computed in a straight forward manner by use of parts F and G of Form 69-9284.

Part F is used to compute the releasable activity at time t_2. To do this, the time t_1 is entered in column 2, the releasable activity (Ci) at time t_1 is entered in column 3, and the time t_2 is entered in column 4. The releasable activity at time t_1 may be the initial releasable activity from Section D, column 5 or the releasable activity computed for some previous time. Next, the elapsed time, t_E is obtained by subtracting t_1 from t_2. The effective cleanup half-life, T_{iCU} from column 9 of Section E is entered in Section F, column 6. The number of elapsed effective cleanup half-lives can now be computed by dividing the elapsed time, T_E by T_{iCU}. These values are entered in column 7 for each isotope and used to determine the cleanup correction factors from Figure 7. These values are entered in column 8. Finally, the releasable activity at time, t_2 is obtained by multiplying the releasable activity at time t_1 by the cleanup correction factor.

The releasable activity is entered in column 2 of part G and the release rate is computed in the same manner as the initial release rate as described above.

b) Effective Clean Half-Lives Changes between t_1 and t_2

If the effective cleanup half-lives change between t_1 and t_2, say at time t_x, then the releasable activity must first be computed at time t_x by the procedure described in section 7.a) above. This releasable activity is then entered in column 3 in a second usage of part F of Form 69-9284. The effective cleanup half-lives for the period between t_x and t_2 are then computed as before using Section E of Form 69-9284. The releasable activity at time, t_2 is then computed as before using Section F of Form 69-9284. Finally, the release rate is computed using Section G of Form 69-9284 as described in Section 5.d.6). above.

8) Total Noble Gas Release Rates

a) Use Table 1 to determine "effective age." Enter value on Work Sheet 69-9284, Section H.2.

b) In column H.3 enter the calculated release rates from Section G of Work Sheet.

c) Sum the release rates and enter result in column H.4.

d) Use Figure 8 to determine the ratio of total noble gas activity to summed Xe-133 and Kr-85 activity. Enter value in column H.5.

e) Multiply the summed release rates (column H.4) times the ratio from column H.5 to determine the noble gas release rate (Ci/sec).

TABLES

1. Initial Effective Age for Various Postulated Accidents.

FIGURES

1. Flow Chart for Determining Release Rate and Total Release.

2. Response of Plant Vent Radiogas Monitors (RE-14).
3. Response of Plant Vent High Range Radioiodine Monitors (RE-29).
4. DBA-LOCA Containment Monitor Exposure Rate Vs. Time Since Accident Onset (RE-30 and 31).
5. DBA-LOCA Noble Gas Release Rate Vs. Time Since Accident Onset (RE-30 and 31).
6. DBA-LOCA I-131 Release Rate Vs. Time Since Accident Onset (RE-30 and 31).
7. Radioactivity Decay Factor Vs. Number of Elapsed Half-Lives.
8. Ratio of Total Noble Gas to Kr-85 and Xe-133 Vs. "Effective Age."

APPENDIX

1. FSAR "Accident Summary Sheets"

ATTACHMENTS

1. Form 69-9260, "Work Sheet for Determination of Release Rate or Total Release from Plant Vent Monitors."
2. Form 69-10555, "Work Sheet for Release Rate Estimation from Containment High Range Area Monitors."
3. Form 69-11105, "Worksheet for Use of Main Steamline Monitors, or RCS Coolant Sample Results During S/G Tube Rupture Accidents".
4. Form 69-9284, "Work Sheet for Estimation of Curie Release."
5. Form 69-10556, "Release Rate Summary."

TABLE 1

INITIAL EFFECTIVE AGE FOR VARIOUS POSTULATED ACCIDENTS

	Initial Effective Age (Hours)
1. Blackout	65
2. Small Loss of Coolant Accident (LOCA)	60
3. Major LOCA (No core melt)	40
4. Major Steamline Break	65
5. Major Feedwater Line Break	65
6. Steam Generator Tube Rupture	65
7. Locked Reactor Coolant Pump (RCP Rotor)	50
8. Fuel Handling Accident in Fuel Handling Building (FHB)	600
9. Control Rod Ejection	40
10. Gas Decay Tank Rupture	80
11. Liquid Holdup Tank Rupture	60
12. Volume Control Tank (VCT) Rupture	60
13. Design Base LOCA with Core Melt	1
14. Gap Activity Release (fresh fuel)	20

NOTE 1: The value tabulated is the initial age assumed for the start of accident; to this must be added any time which has passed since the start of the accident. In the case of the fuel handling accident in the Fuel Handling Building, the FSAR assumes 600 hour (25 day) cooling. If the time since the damaged fuel assembly has operated is longer than 25 days, use this actual cooling time as the effective age.

NOTE 2: An "effective age" of 0 hours corresponds to the mixture which is contained in the fuel pellets at the time of reactor shutdown.

FIGURE 1

FLOW CHART FOR DETERMINING RELEASE RATE AND TOTAL RELEASE

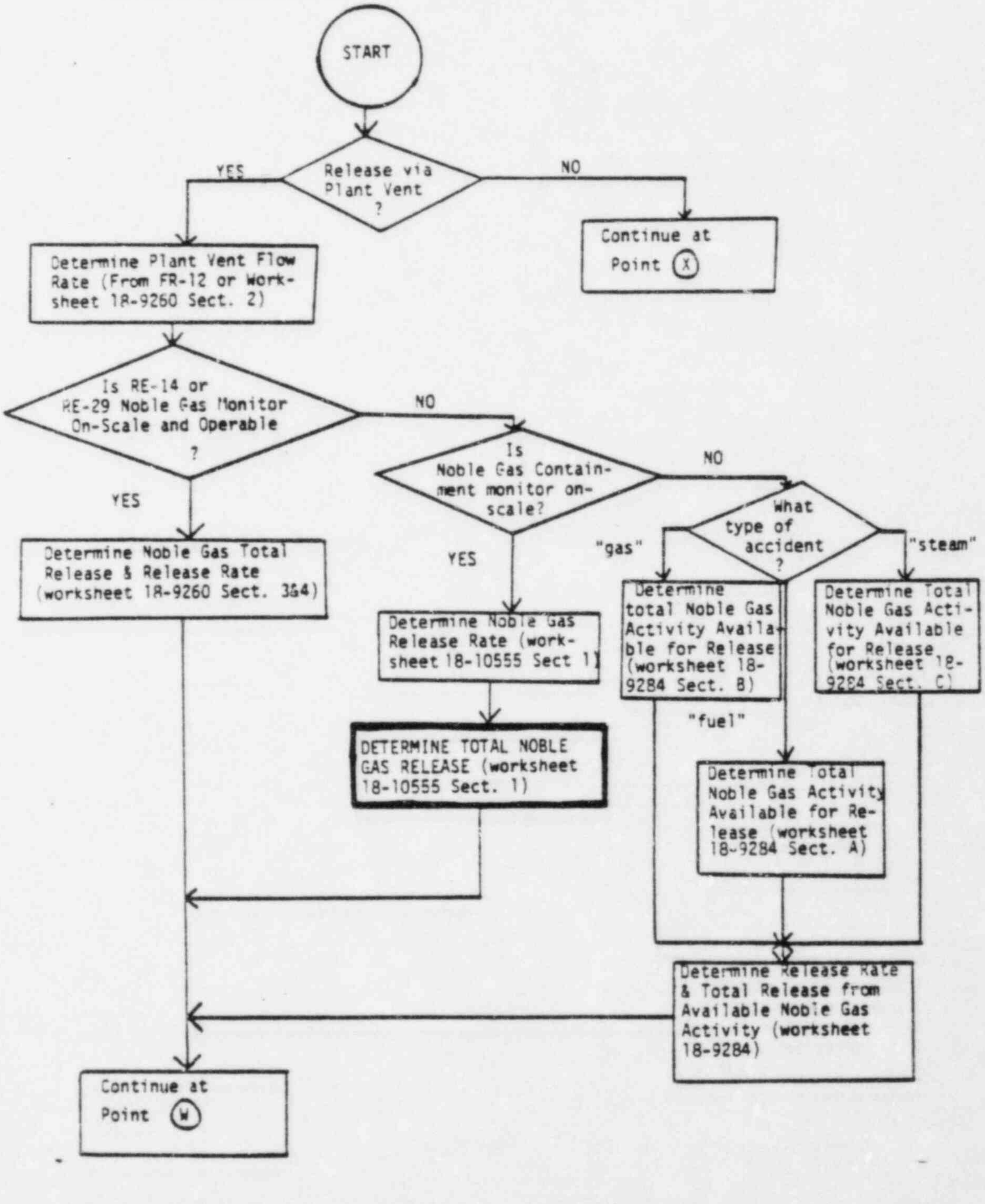


FIGURE 1 (continued)

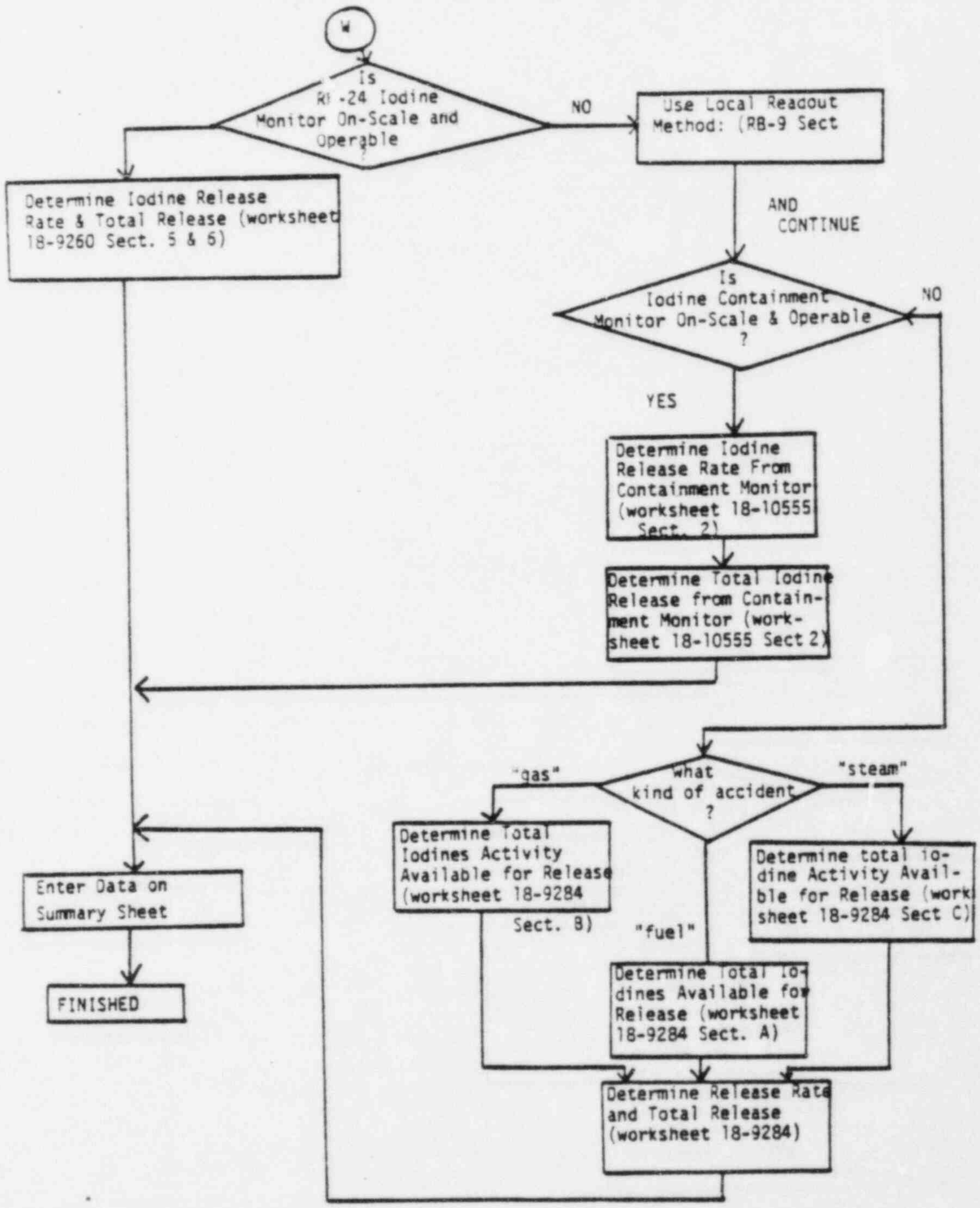


FIGURE 1 (continued)

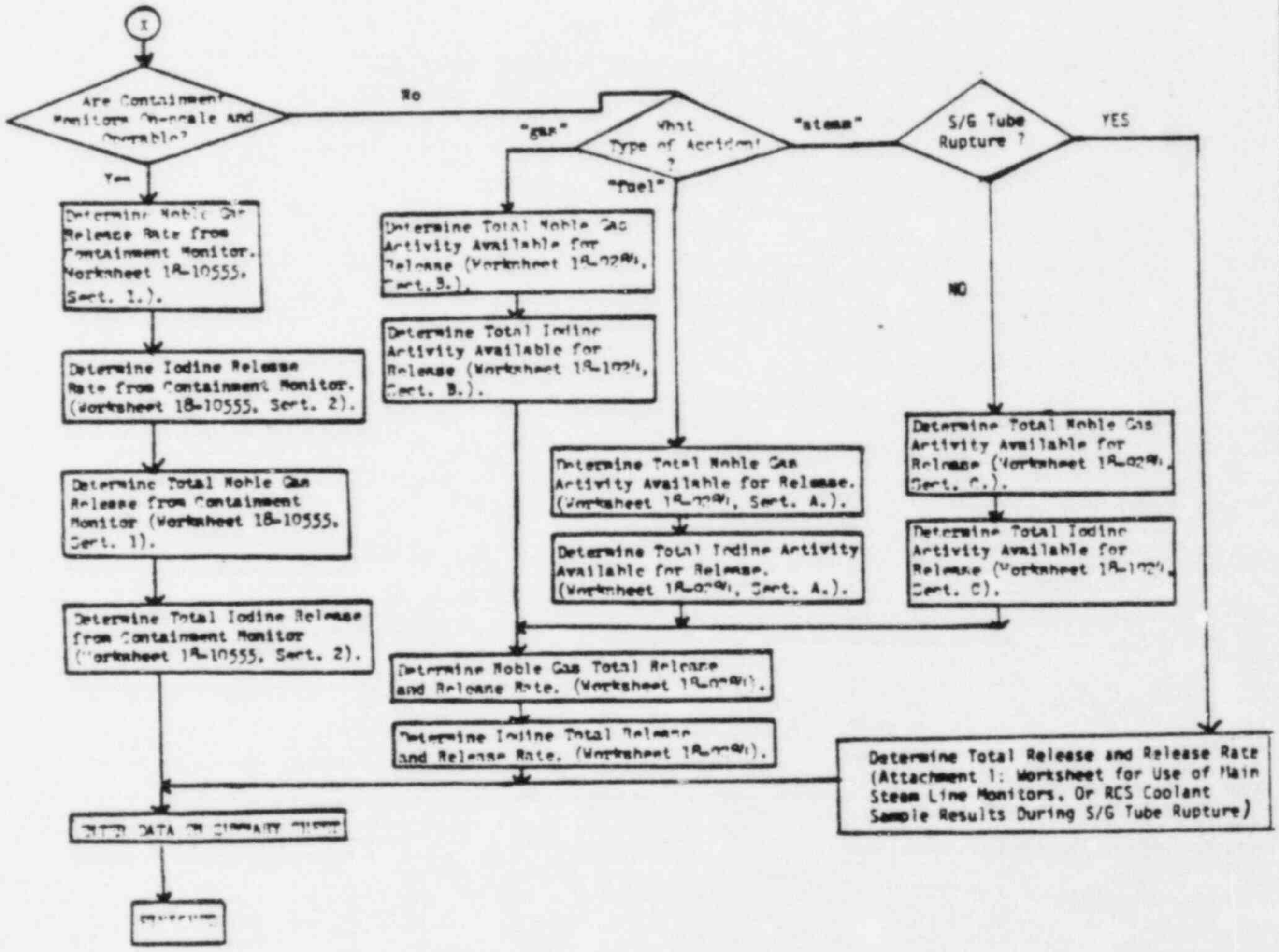


FIGURE 2

RESPONSE OF PLANT VENT RADIOGAS MONITOR

(RE-14)

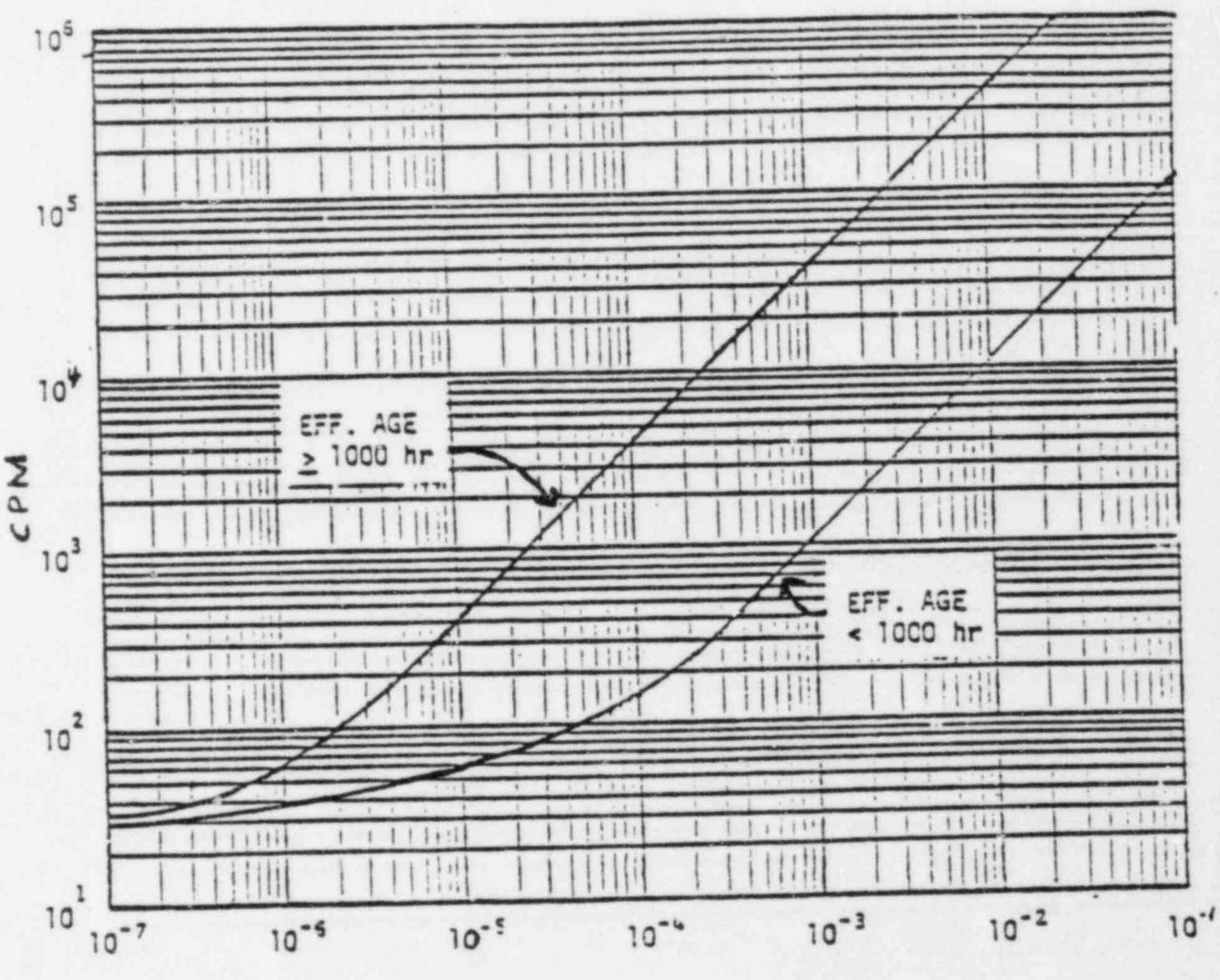


FIGURE 3

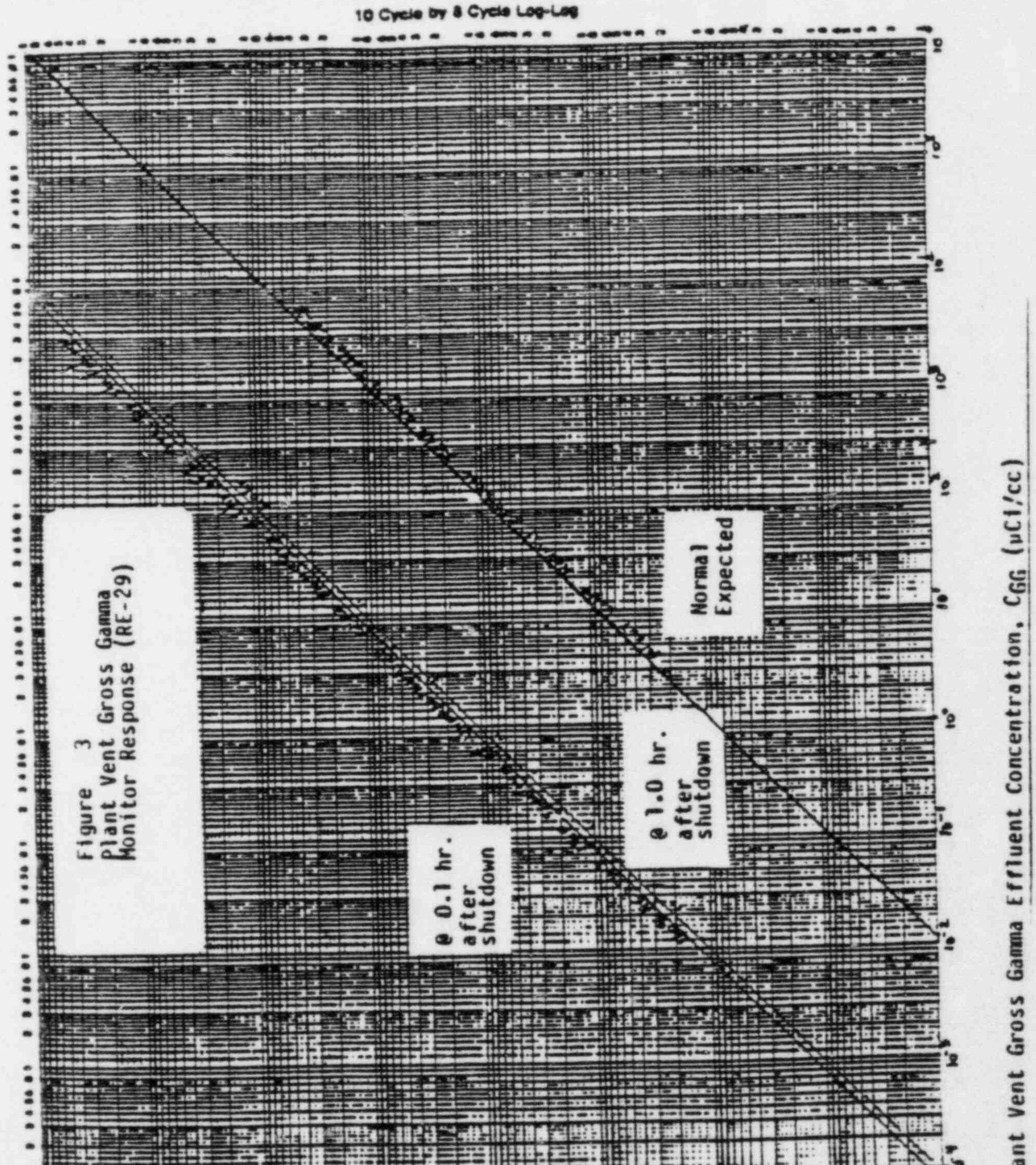

10 Cycle by 8 Cycle Log-Log
Figure 3
Plant Vent Gross Gamma
Monitor Response (RE-29)
Normal
Expected
@ 1.0 hr.
after
shutdown
@ 0.1 hr.
after
shutdown
Plant Vent Gross Gamma Effluent Concentration, C_{GG} (µCi/cc)
Net Gross Gamma Monitor Response (mR/hr)

FIGURE 4
DBA-LOCA CONTAINMENT MONITOR EXPOSURE RATE
VS.
TIME SINCE ACCIDENT ONSET (RE-30 AND 31)

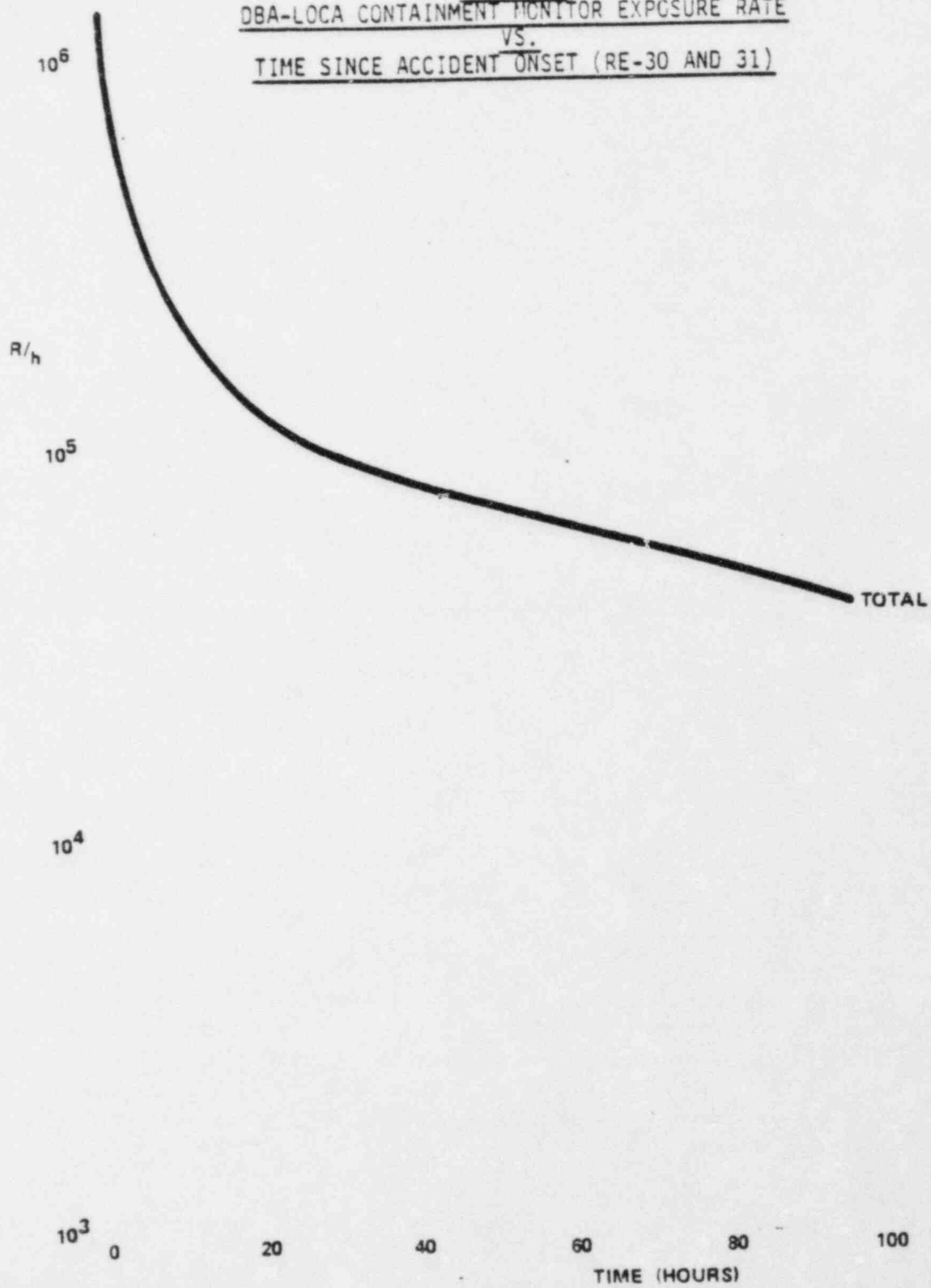
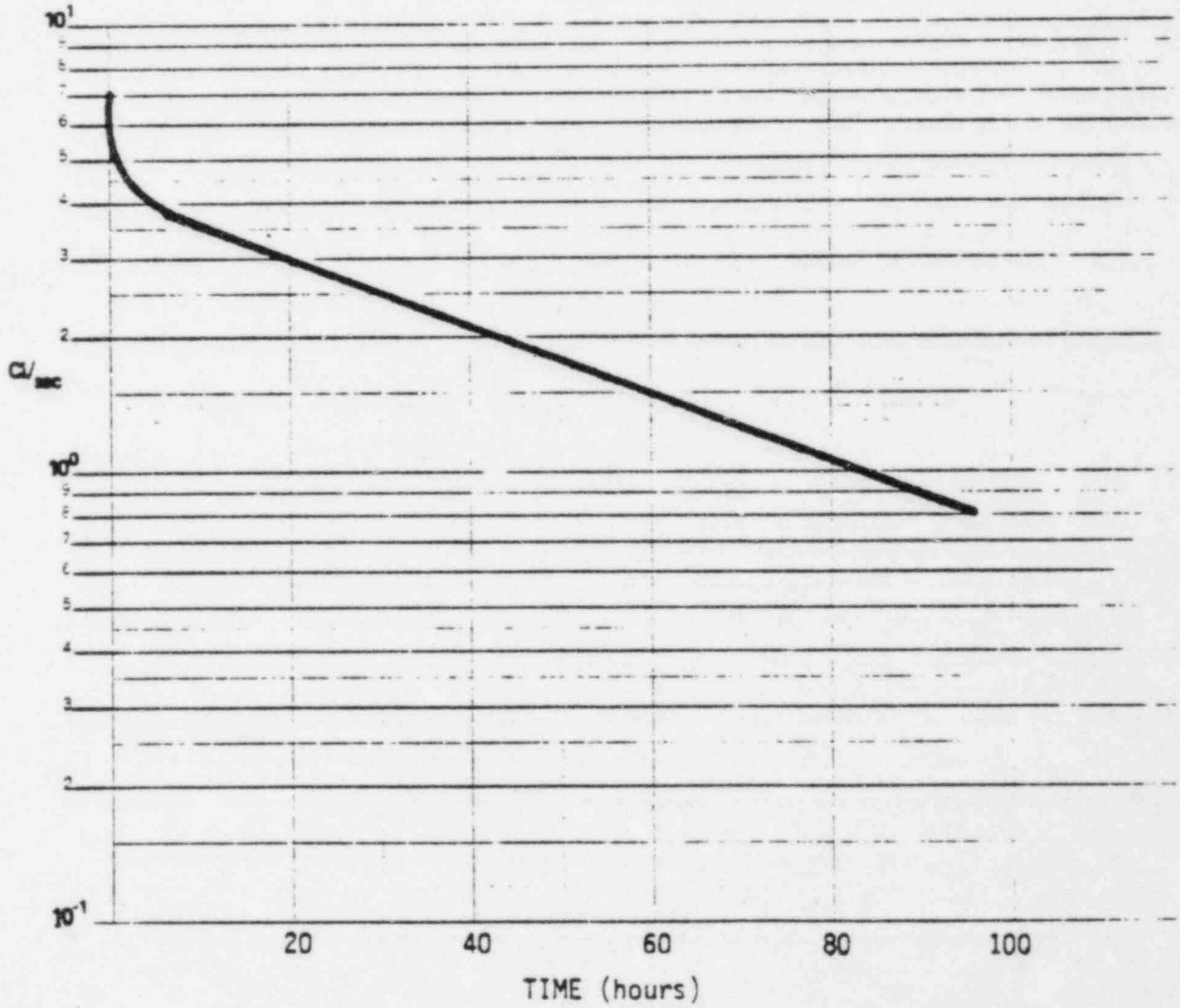


FIGURE 5

DBA-LOCA NOBLE GAS RELEASE RATE VS. TIME SINCE ACCIDENT ONSET (RE-30 and 31)



DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2	NUMBER EP RB-9 REVISION 2 DATE 11/15/83 PAGE 32 OF 61
TITLE CALCULATION OF RELEASE RATE AND INTEGRATED RELEASE	

FIGURE 6

DBA-LOCA I-131 RELEASE RATE VS. TIME SINCE ACCIDENT ONSET (RE-30 and 31)

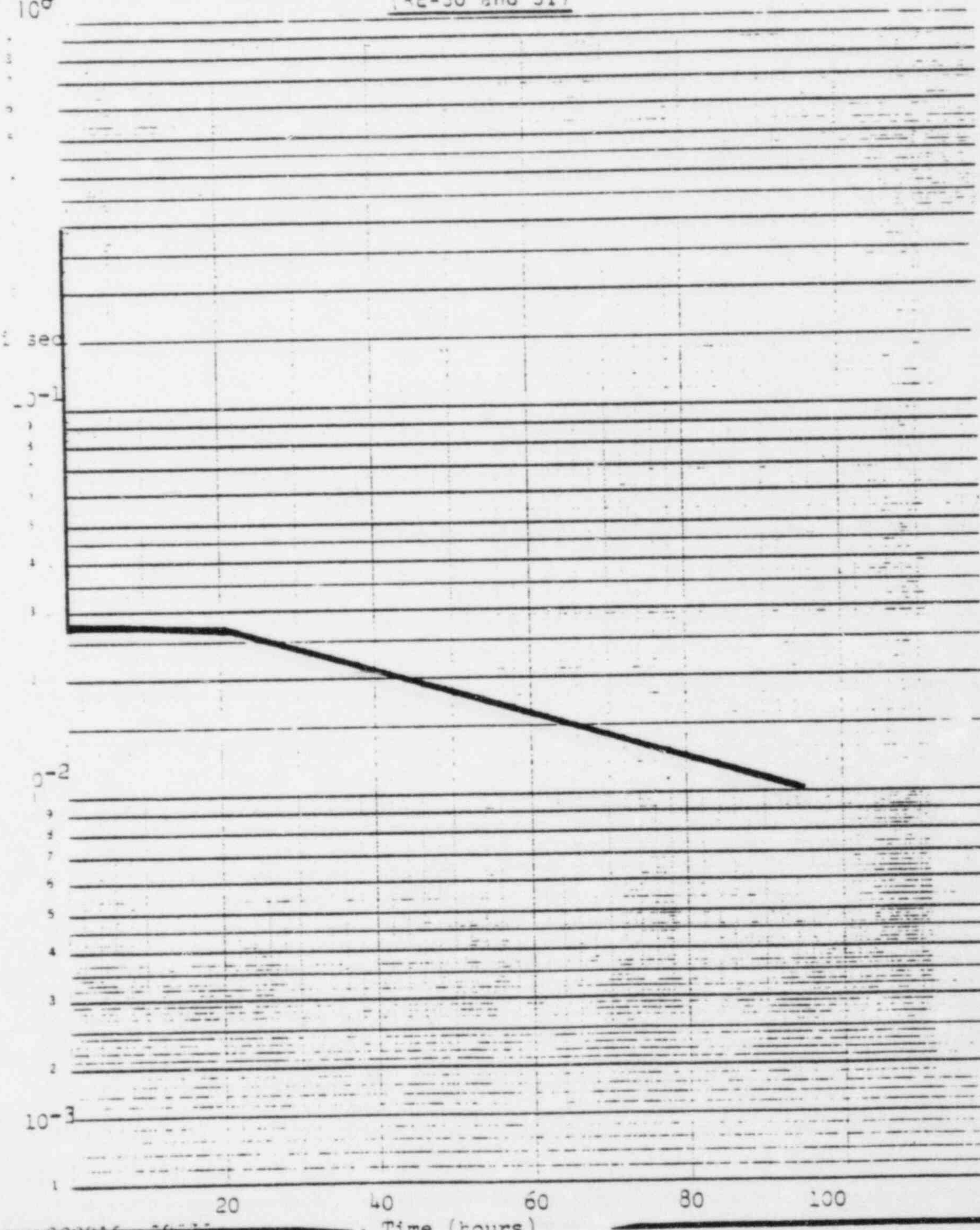


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FIGURE 7

RADIOACTIVITY DECAY FACTOR VS. NUMBER OF ELAPSED HALF-LIVES

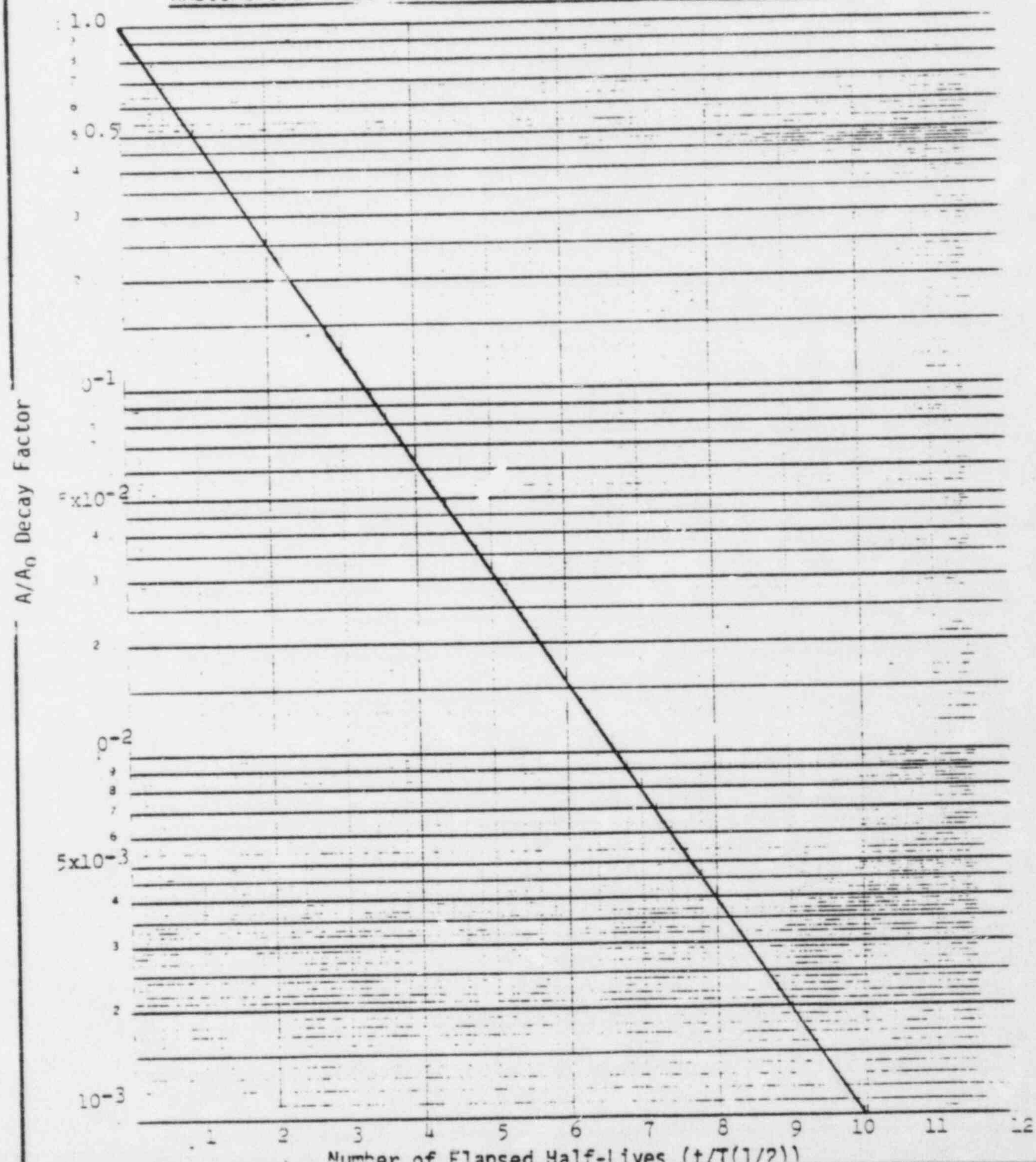


FIGURE 7 (continued)

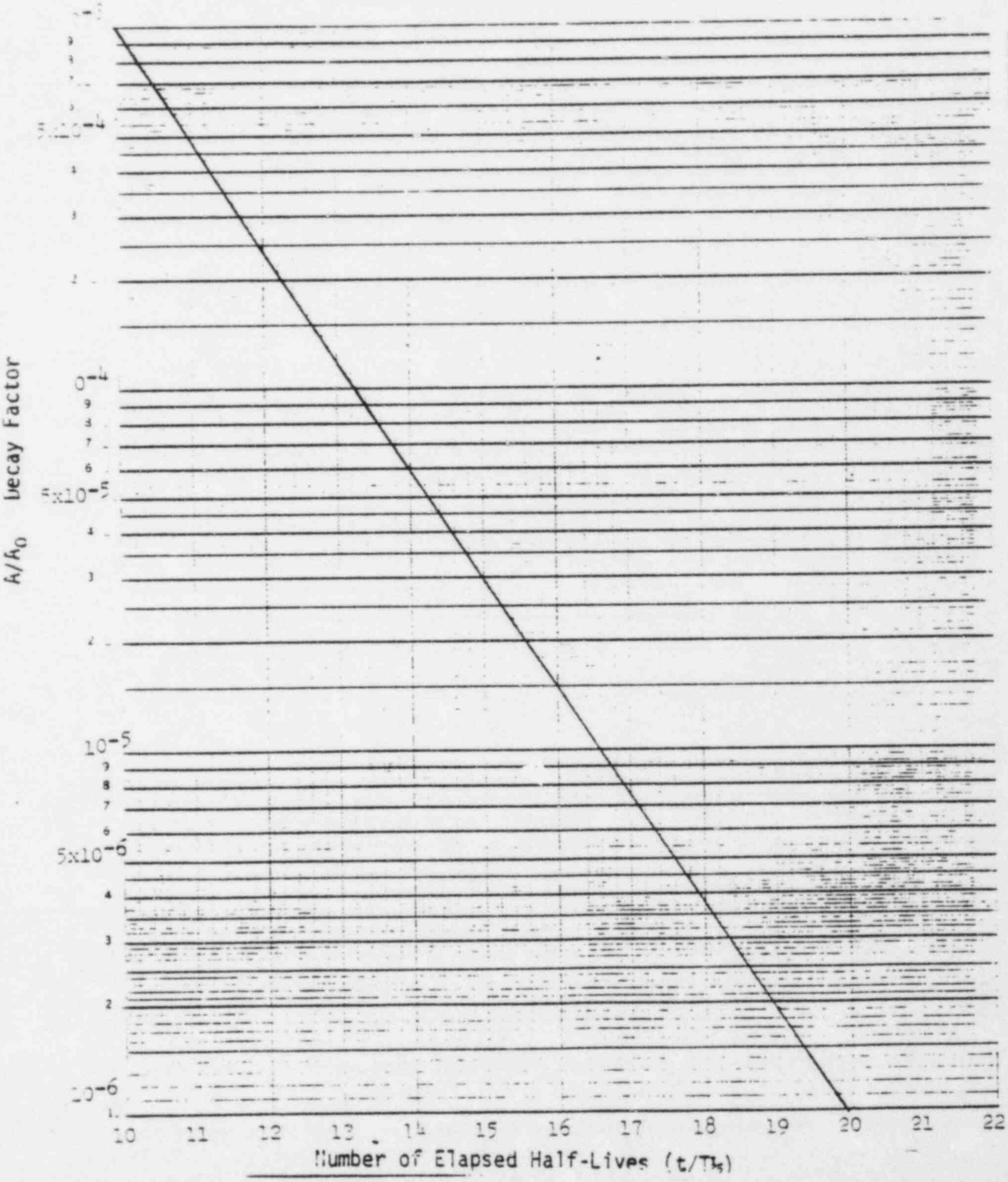
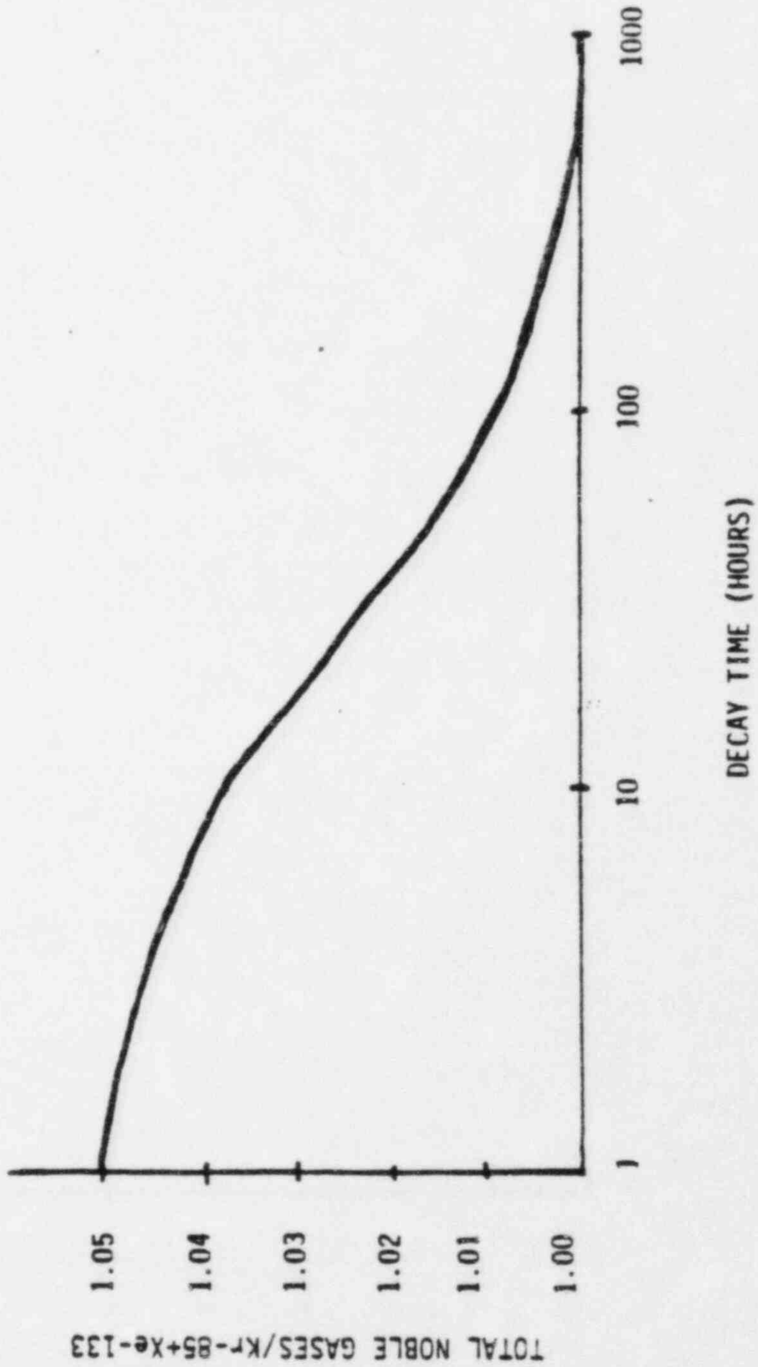


FIGURE 8

RATIO OF TOTAL NOBLE GAS TO KR-85 AND XE-133

VS. "EFFECTIVE AGE"



APPENDIX 1

ACCIDENT SUMMARY SHEETS

This attachment contains summary sheets for the various postulated accidents which have been analyzed in the FSAR. These sheets contain both the "design basis" and "expected" case variables which were assumed in the FSAR analyses. The sheets can be used to compare actual measurements with assumed numbers from the FSAR, in order to help evaluate how things are going in relation to predictions, or they can be used as a source of data to supply unavailable numbers in calculations which are performed at the time of the accident.

Two sets of data are included. The "design basis" case is expected to be highly conservative, where every variable is at a worst-case condition. The "expected" case is the best estimated prediction of what might actually occur. When FSAR values are used to make calculations or predictions at the time of the accident, the "design basis" values can be used to provide a quick upper limit result, but as soon as data becomes available which tends to confirm one case or the other, the one which best agrees with the data should be used.

The accident classifications identified in this attachment are based on the activity releases. Other emergency procedures may have different classifications which are based on the initiating event.

The summary sheets provided are:

A. 1A MAJOR LOCA
B. 1B MAJOR STEAM LINE BREAK
C. 1C MAJOR FEEDWATER LINE BREAK
D. 1D BLACKOUT (OR PLANT COOLDOWN WITH ATMOSPHERIC DUMP)
E. 1E SMALL LOCA
F. 1F TUBE RUPTURE
G. 1G LOCKED ROTOR
H. 1H FUEL HANDLING ACCIDENT INFUEL HANDLING BUILDING
I. 1I FUEL HANDLING ACCIDENT IN CONTAINMENT
J. 1J ROD EJECTION ACCIDENT
K. 1K GAS DECAY TANK RUPTURE
L. 1L LIQUID HOLDUP TANK RUPTURE
M. 1M VCT RUPTURE

SUMMARY SHEET 1A
MAJOR LOCA

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Total Release to Containment Free Volume, Ci			
a. Xe-133	2.03×10^8	1.36×10^6	
b. Other Noble Gases	5.73×10^8	4.27×10^5	
c. I-131	2.21×10^7	1.82×10^5	
d. Other Iodine	1.90×10^8	2.73×10^5	
e. Effective Age of Mixture (hr)	0	20	
f. Release Assumption	100% of core N.G., 25% of core iodines	100% of gap N.G., 25% of gap iodines	
2. Containment Spray Effectiveness			
a. Removal half-life (hrs)	0.022	0.0075	
b. Number of operable spray pumps	1	2	
3. Containment Leak Rate (%/day)	0.1 for 1st day, 0.05 after 1st day	0.05 for 1st day, 0.025 after 1st day	

SUMMARY SHEET 1A (Continued)

4.	Total Release to Environs, First 2 Hours, Ci		
	a. Xe-133	16,840	56
	b. Other Noble Gases	25,930	21
	c. I-131	191	0.05
	d. Other Iodine	1,325	0.08
	e. Effective Age of Mixture	1	40
	f. Release Mechanism	Containment Leakage	Containment Leakage
5.	$(\chi/\dot{Q})$ CL (sec/m^3)		
	a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}
	b, 100000m (6 mi. LPZ)	2.20x20^{-5}	2.20×10^{-6}
6.	Whole Body Dose Results		
	a. Total 800m dose for 1st two hours (mR)	5,600	0.365
	b. Total 10000m dose for 30 days (mR)	567	0.06
7.	Thyroid Dose Results		
	a. Total 800m dose for 1st two hours (mR)	95,900	1.25
8.	Accident Classification	General Emergency	Alert
9.	Miscellaneous		
	a. Containment-free volume cc	7.36×10^{10}	
	b. RCS Coolant Mass (gm)	2.4×10^{8}	

SUMMARY SHEET 1B
MAJOR STEAM LINE BREAK

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Initial Conditions and Assumptions			
a. Primary Coolant Activity (μCi/gm)			
1) Xe-133	270	67.2	
2) I-131	2.6	0.65	
3) Other Iodine	7.9	2.0	
b. Secondary Water Activity (μCi/gm)			
1) I-131	0.015	0.44×10^{-4}	
2) Other Iodines	0.037	0.90×10^{-4}	
c. Assumed Fuel Defects (%)	1	0.2	
d. Primary to Secondary Leakage (gpm)	1	0.014	
e. Steam Release, 1st Two Hours (lbs)			
1) Failed Generator	97,000		
2) Other generator (atmospheric dump)	520,000		
f. Total Steam Release During 8-Hour Cooldown (lbs)	1,600,000		
g. Liquid Release Fraction for Iodine			
1) Failed Generator	0.1		
2) Other generators	0.01		

SUMMARY SHEET 1B (Continued)

	PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
2.	Activity Release to Environs, First 2 Hours (Ci)			
	a. Xe-133	56.8	0.172	
	b. Other Noble Gases	5.2	0.016	
	c. I-131	0.157	0.00045	
	d. Other Iodines	0.047	0.0013	
	e. Effective Age of Mixture (hrs)	65	65	
3.	$(\chi/\dot{Q})$ CL (sec/m^3)			
	a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
	b. 10000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-6}	
4.	Whole Body Dose Results			
	a. Total 800m dose for 1st two hours (mR)	1.8	0.0006	
	b. Total 10000m dose for 30 days (mR)	0.03	0.0010	
5.	Thyroid Dose Results			
	a. Total 800m dose for 1st two hours (mR)	65	0.012	
	b. Total 10000m dose for 30 days (mR)	66	0.012	
6.	Accident Classification	Alert	Alert	

SUMMARY SHEET 1B (Continued)

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
7. Miscellaneous			
a. Fluid Mass/Stm Gen (lbs)			
1) Water	95,100		
2) Steam	6,620		
b. Safety Valve and Steam Dump Valve Capacities (lb/hr/valve)			
1) S.G. safety valve	800,000		
2) 10% atmospheric dump	380,000		
3) 35% atmospheric dump	597,000		

SUMMARY SHEET 1C
MAJOR FEEDWATER LINE BREAK

The release from this accident comes from release of steam by safety valves and/or atmospheric steam dump of steam generator water during cooldown if the condenser is not available. The steam generator water is contaminated if there is tube leakage. The feedwater itself which is released has very little activity in it and is ignored. This accident is basically the same as a steam-line break and summary sheet 9B can be used. Note, however, that the steam release will be through relief valves and so the iodine liquid release fraction should be 0.01 for the entire release. This will reduce the thyroid dose somewhat from the steam-line break case.

SUMMARY SHEET 1D
BLACKOUT (PLANT COOLDOWN WITH ATMOSPHERIC DUMP)

The release from this accident comes from release of steam by safety valves and/or atmospheric steam dump of steam generator water is contaminated if there is tube leakage. This accident is basically the same as a steam-line break and summary sheet 9B can be used. Note, however, that the steam release will be through relief valves and so the iodine liquid release fraction should be 0.01 for the entire release. This will reduce the thyroid dose somewhat from the steam-line break case.

SUMMARY SHEET 1E
SMALL LOCA (RELEASE OF COOLANT TO CONTAINMENT)

	PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1.	Initial Coolant Activity (μCi/gm)			
	a. Xe-133	270	45.7	
	b. Other Noble Gases	30	5.6	
	c. I-131	2.62	0.45	
	d. Other Iodine	7.88	1.35	
	e. Effective Age of Mixture (hr)	60	60	
	f. Fuel Defects (%)	1	0.2	
2.	Initial Release to Containment (Ci)			
	a. Xe-133	65,430	16,280	
	b. Other Noble Gases	7,950	1,980	
	c. I-131	63	16	
	d. Other Iodine	193	48	
	e. Assumption	100% of Coolant N.G. activity +10% of coolant iodines	100% of Coolant N.G. Activity +10% of coolant iodines	
3.	Containment Spray Effectiveness			
	a. Removal Half-life (hrs)	0.022	0.0075	

SUMMARY SHEET 1E (Continued)

	PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
	b. Number of operable spray pumps	1	2	
	c. Containment Leak Rate (%/day)	0.1 for 1st day, 0.05 after 1st day	0.05 for 1st day, 0.025 after 1st day	
4.	Containment Leak Rate (%/day)	0.1	0.05	
5.	$(\chi/\dot{Q})$ CL (sec/m^3)			
	a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
	b. 10000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-6}	
6.	Whole Body Dose Results			
	a. Total 800m dose for 1st two hours (mR)	0.18	0.004	
	b. Total 10000m dose for 30 days (mR)	0.05	0.001	
7.	Thyroid Dose Results			
	a. Total 800m dose for 1st two hours (mR)	0.2	0.0009	
	b. Total 10000m dose for 30 days (mR)	0.03	0.0001	
8.	Accident Classification	Alert	Alert	

SUMMARY SHEET 1E (Continued)

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
9. Miscellaneous			
a. Containment-Free Volume (cc)	7.36×10^{10}		
b. RCS Coolant Mass (gm)	2.4×10^{8}		
c. Liquid Release Fraction for Iodine	0.1		

SUMMARY SHEET 1F
TUBE RUPTURE

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Initial Conditions and Assumptions			
a. Primary Coolant Activity (μCi/gm)			
1) Xe-133	270	67.2	
2) I-131	2.6	0.65	
3) Other Iodine	7.9	2.0	
b. Secondary Water Activity (μCi/gm)			
1) I-131	0.015	0.44×10^{-4}	
2) Other Iodines	0.037	0.90×10^{-4}	
c. Assumed Fuel Defects (%)	1	0.2	
d. Primary to Secondary Leakage (gpm)	1	0.014	
e. Steam Release, 1st Two Hours (lbs)			
1) Failed generator	31,000		
2) Other generators (atmospheric dump)	380,000		
f. Total Steam Release During 8 hour Cooldown (lbs)	1,600,000		
g. Liquid Release Fraction for Iodine			
1) Failed generator	0.01		
2) Other generators	0.01		

SUMMARY SHEET 1F (Continued)

	PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
2.	Total Release to Environs First 2 hours (Ci)			
	a. Xe-133	10,980	2,383	
	b. Other Noble Gases	1,067	234	
	c. I-131	0.75	0.14	
	d. Other Iodines	3.1	0.62	
	e. Effective Age of Mixture (hrs)	65	65	
3.	$(\chi/\dot{Q})$ CL (sec/m^3)			
	a. 800 m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
	b. 10000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-6}	
4.	Whole Body Dose Results			
	a. Total 800m dose for 1st two hours (mR)	360	7.7	
	b. Total 10000m dose for 30 days (mR)	15	0.3	
5.	Thyroid Dose Results			
	a. Total 800m dose for 1st two hours (mR)	340	4.3	
	b. Total 10000m dose for 30 days (mR)	15	0.2	
6.	Accident Classification	Alert	Alert	

SUMMARY SHEET 1F (Continued)

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
7. Miscellaneous			
a. Fluid Mass/Stem Gen. (lbs)			
1) Water	95,000		
2) Steam	6,620		
b. Safety Valve and Steam Dump Valve Capacities (lbs/hr/valve)			
1) S.G. safety valve	800,000		
2) 10% atmospheric dump	380,000		
3) 35% atmospheric dump	597,000		

SUMMARY SHEET 1G
LOCKED ROTOR ACCIDENT

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Total Release to Environs, 1st Two Hours (Ci)			
a. Xe-133	97	0.73	
b. Other Noble Gases	19.6	0.21	
c. I-131	0.24	0.003	
d. Other Iodines	0.36	0.003	
e. Effective Age of Mixture	50	50	
f. Assumptions			
1) Coolant Activity	1% fuel defects +3% of gap activity	0.2% fuel defects +3% of gap activity	
2) Primary to Secondary Leakage (gpm)	1	0.014	
3) Secondary Steam Release, 1st Two Hours (lbs)	617,000	617,000	
4) Total Steam Release During 8 Hour Cooldown (lbs)	1,600,000	1,600,000	
2. $(\chi/\dot{Q})$ CL (sec/m^3)			
a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
b. 1000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-5}	

SUMMARY SHEET 1G (Continued)

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
3. Whole Body Dose Results			
a. Total 800m dose for 1st two hours (mR)	4.4	0.004	
b. Total 10000m dose for 30 days (mR)	0.5	0.0004	
4. Thyroid Dose Results			
a. Total 800m dose for 1st two hours (mR)	82	0.06	
b. Total 10000m dose for 30 days (mR)	27	0.02	
5. Accident Classification	Alert	Alert	
6. Miscellaneous			
a. Fluid Mass/Stm Gen. (lbs)			
1) Water		95,100	
2) Steam		6,620	
b. Safety Valve and Steam Dump Valve Capacity (lbs/hr/valve)			
1) S.G. safety valve		800,000	
2) 10% atmospheric dump		380,000	
3) 35% atmospheric dump		597,000	
c. Liquid Release Fraction for Iodines	0.01		

SUMMARY SHEET 1H
FUEL HANDLING ACCIDENT IN FUEL HANDLING BLDG

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Initial Conditions			
a. Radial Peaking Factor of Damaged Assembly	1.65	1.26	
b. Elapsed Time Since Reactor Shutdown (hrs)	100	100	
c. Type of Release to Pool	100% of assembly gap activity	100% of assembly gap activity	
d. Bundle Submergence (ft)	26	26	
e. Pool Decontamination Factor for Iodine	100	760	
f. Total Assembly Gap Activity at Time of Accident			
1) Xe-133	100,000	8,137	
2) Other Noble Gases	4,500	1,500	
3) I-131	52,670	5,282	
4) Other Iodines	7,000	220	
5) Effective Age of Mixture (hr)	600	600	
2. $(\chi/\dot{Q})$ CL (sec/m^3)			
a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
b. 1000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-6}	
3. Total Release to Environs, 1st Two Hours (Ci)			
a. Xe-133	100,400	523	
b. Other Noble Gases	4,100	101	

SUMMARY SHEET 1H (Continued)
FUEL HANDLING ACCIDENT IN FUEL HANDLING BLDG

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
c. I-131	80	0.005	
d. Other Iodines	10	0.0002	
e. Effective Age of Mixture (hrs)	600	600	
4. Whole Body Dose Results			
a. Total 800m dose for 1st two hours (mR)	2,450	1.5	
b. Total 10000m dose for 30 days (mR)	102	0.06	
5. Thyroid Dose Results			
a. Total 800m dose for 1st two hours (mR)	22,200	0.08	
b. Total 10000m dose for 30 days (mR)	923	0.003	
6. Accident Classification	Site Emergency	Alert	
7. Miscellaneous			
a. Fuel Handling Building Volume (ft^3)	435,000		
b. Fuel Handling Building Exhaust Rate (cfm)	35,700	35,700	
c. Filter Cleanup Factor	0.10	0.01	

SUMMARY SHEET II
FUEL HANDLING ACCIDENT IN CONTAINMENT

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Initial Conditions			
a. Radial Peaking Factor of Damaged Assembly	1.65	1.26	
b. Elapsed Time Since Reactor Shutdown (hrs)	100	100	
c. Type of Release to Pool	100% of assembly gap activity	100% of assembly gap activity	
d. Bundle Submergence (ft)	26	26	
e. Pool Decontamination Factor for Iodine	100	760	
f. Total Assembly Gap Activity at Time of Accident (Ci)			
1) Xe133	100,000	8,137	
2) Other Noble Gases	4,500	1,500	
3) I-131	52,670	5,282	
4) Other Iodines	7,000	220	
5) Effective Age of Mixture (hrs)	600	600	
2. $(\chi/\dot{Q})$ CL (sec/m^3)			
a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
b. 1000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-5}	
3. Total Release to Environs, 1st Two Hours (Ci)			
a. Xe-133	12,460	38	

SUMMARY SHEET II (Continued)
FUEL HANDLING ACCIDENT IN CONTAINMENT

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
b. Other Noble Gases	557	7	
c. I-131	65	0.033	
d. Other Iodines	8.7	0.0013	
e. Effective Age of Mixture (hrs)	600	600	
4. Whole Body Dose Results			
a. Total 800m dose for 1st two hours (mR)	0.31	0.0001	
b. Total 10000m dose for 30 days (mR)	0.013	$4x10^{-6}$	
5. Thyroid Dose Results			
a. Total 800 m dose for 1st two hours (mR)	18.4	$6x10^{-4}$	
b. Total 10000m dose for 30 days (mR)	0.76	$3x10^{-5}$	
6. Accident Classification	Site Emergency	Alert	
7. Miscellaneous Activity Release Mechanism	Activity released from cavity to containment atmosphere is confined directly above the cavity water level. It is picked up by the fan coolers and sent out through the containment purge.		

SUMMARY SHEET 1J
ROD EJECTION ACCIDENT

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Total Release to Containment Free Volume (Ci)			
a. Xe-133	2.01×10^{5}	1.52×10^{5}	
b. Other Noble Gases	6.82×10^{4}	6.22×10^{4}	
c. I-131	7.32×10^{3}	7.28×10^{3}	
d. Other Iodine	1.11×10^{4}	1.09×10^{4}	
e. Effective Age of Mixture (hrs)	40	40	
f. Release Assumption	Coolant activity (1% defects) plus 10% of core gap activity times a liquid release fraction of either 0.1 (for I) or 1.0 (for N.G.)	Coolant activity (0.2% defects) plus 10% of core gap activity times a liquid release fraction of either 0.1 (for I) or 1.0 (for N.G.)	
2. Containment Spray Effectiveness			
a. Removal half-life (hrs)	0.022	0.0075	
b. Number of operable spray pumps	1	2	
3. Containment Leak Rate (%/day)	0.1	0.05	
4. $(\chi/\dot{Q})$ CL (sec/m^3)			
a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
b. 1000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-5}	

SUMMARY SHEET 1J (Continued)
ROD EJECTION ACCIDENT

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
5. Total Release to Environs, 1st 2 Hours (Ci)			
a. Xe-133	11.2	5.6	
b. Other Noble Gases	4.1	2.0	
c. I-131	0.0098	0.002	
d. Other Iodine	0.015	0.002	
e. Effective Age of Mixture (hrs)	40	40	
f. Release Mechanism	Containment Leakage	Containment Leakage	
6. Whole Body Dose Results			
a. Total 800m dose for 1st two hours (mR)	0.73	0.04	
b. Total 1000m dose for 30 days (mR)	0.13	0.006	
7. Thyroid Dose Results			
a. Total 800m dose for 1st two hours (mR)	3.3	0.04	
b. Total 10000m dose for 30 days (mR)	0.14	0.002	
8. Accident Classification	Alert	Alert	
9. Miscellaneous			
a. Containment free volume (cc)	7.36×10^{10}		
b. RCS Coolant Mass (gm)	2.4×10^{8}		

SUMMARY SHEET 1K
GAS DECAY TANK RUPTURE

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Total Release to Environs, 1st Two Hours (Ci)			
a. Xe-133	65,400	16,300	
b. Other Noble Gases	7,300	2,140	
2. $(\chi/\mathring{Q})$ CL (sec/m^3)			
a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
b. 10000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-5}	
3. Whole Body Dose Results			
a. Total 800m dose for 1st two hours (mR)	2,010	44	
b. Total 10000m dose for 30 days (mR)	84	2	
4. Accident Classification	Site Emergency	Alert	
5. Miscellaneous			
a. Tank Volume (cc)	2.18×10^{-7}		
b. Tank Press	100 psi		
c. Volume Released (cc)	1.48×10^{8}cc		

SUMMARY SHEET 1L
LIQUID HOLDUP TANK RUPTURE

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Activity in Holdup Tank (Ci)			
a. Xe-133	51,000	10,200	
b. Other Noble Gases	4,710	930	
c. I-131	492	98.3	
d. Other Iodines	1,086	217	
e. Effective Age of Mixture (hrs)	60	60	
2. Cleanup Parameters			
a. Liquid Release Fraction for Iodines from Tank to Auxiliary Building Atmosphere	10^{-4}	10^{-4}	
b. Charcoal Filter Cleanup Factor	0.1	0.01	
c. Release Duration (hrs)	2	2	
3. Activity Release to Environs, 1st Two Hours (Ci)			
a. Xe-133	51,000	10,200	
b. Other Noble Gases	4,710	930	
c. I-131	0.00492	0.0098	
d. Other Iodines	0.01086	0.00217	
4. $(\chi/\dot{Q})$ CL (sec/m^3)			
a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
b. 1000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-5}	

SUMMARY SHEET 1L (Continued)
LIQUID HOLDUP TANK RUPTURE

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
5. Whole Body Dose Results			
a. Total 800m dose for 1st two hours (mR)	1,440	37	
b. Total 10000m dose for 30 days (mR)	60	1.6	
6. Thyroid Dose Results			
a. Total 800m dose for 1st two hours (mR)	1.93	0.003	
b. Total 10000m dose for 30 days (mR)	0.08	0.0001	
7. Accident Classification	Site Emergency	Alert	
8. Miscellaneous			
a. Tank Volume (cc)		3.03×10^8	

SUMMARY SHEET 1M
VOLUME CONTROL TANK RUPTURE

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
1. Activity in VCT (Ci)			
a. Xe-133	3,330	828	
b. Other Noble Gases	198	42	
c. I-131	12.1	3.0	
d. Other Iodines	35	8.7	
e. Effective Age of Mixture (hrs)	60	60	
2. Cleanup Parameters			
a. Liquid Release Fraction for Iodines from Tank to Auxiliary Building Atmosphere	10^{-4}	10^{-4}	
b. Charcoal Filter Cleanup Factor	0.1	0.01	
c. Release Duration (hrs)	2	2	
3. Activity Release to Environs, 1st Two Hours (Ci)			
a. Xe-133	3,330	828	
b. Other Noble Gases	198	42	
c. I-131	0.00012	0.000003	
d. Other Iodines	0.00035	0.00009	

SUMMARY SHEET 1M (Continued)
VOLUME CONTROL TANK RUPTURE

PARAMETER	FSAR DBA	FSAR EXPECTED	ACTUAL
4. $(\chi/\dot{Q})$ CL (sec/m^3)			
a. 800m (site boundary)	5.29×10^{-4}	5.29×10^{-5}	
b. 1000m (6 mi. LPZ)	2.20×10^{-5}	2.20×10^{-5}	
5. Whole Body Dose Results			
a. Total 800m dose for 1st two hours (mR)	465	9.3	
b. Total 10000m dose for 30 days (mR)	19	0.4	
6. Thyroid Dose Results			
a. Total 800m dose for 1st two hours (mR)	0.03	0.00004	
b. Total 10000m dose for 30 days (mR)	0.001	0.000001	
7. Accident Classification	Site Emergency	Alert	
8. Miscellaneous			
a. Tank Volume (cc)	1.1×10^7		

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

TITLE: WORK SHEET FOR DETERMINATION OF RELEASE RATE OR TOTAL RELEASE FROM PLANT VENT MONITORS - EP RB-9

1. Basic Accident Information

Unit ______ Date/Time of Occurence ___________/___________

Description ____________________________ Init. Eff. Age (hr)___________

2. Determine Plant Vent Flow Rate (Affected Unit Only)

a. Ventilation equipment in operation (affected unit only)

____________FHB exhaust fans @ 35750 cfm/fan = ________________(cfm)

____________Aux. Bldg. exhaust fans @ 73500 cfm/fan = __________(cfm)

____________Cont. purge exhaust fans @ 55000 cfm /fan = _________(cfm)

____________Cont. H_2 purge fan @ 300 cfm/fan = ________________(cfm)

b. Total plant vent flow rate (sum above or read directly from FR-12 in Control Room) = $\boxed{\quad\quad}$ (cfm)

c. Conversion to cc/sec

____________ (cfm) x 472 = $\boxed{\quad\quad}$ (cc/sec)

3. Determine Noble Gas Release Rate (RE-14 and RE-29)

RE	READING	① Vent Concentration (μCi/cc)	② (Vent Flow) (cc/sec)	③ Conversion (Ci/μCi)	①②③ $\dot{Q}$ (Ci/sec)
14	______cpm	__________	__________	10^{-6}	$\boxed{\quad}$
29	______mR/hr	__________	__________	10^{-6}	$\boxed{\quad}$

4. Estimating Total Curie Noble Gas Release

RE	① $\dot{Q}$ (Ci/sec)	② Δt (sec)	① x ② Q(Ci)	
14	______	______	______	Sum of Q (RE-14)
	______	______	______	$\boxed{\quad}$
	______	______	______	
29	______	______	______	Sum of Q (RE-29)
	______	______	______	$\boxed{\quad}$
	______	______	______	

TITLE: WORK SHEET FOR DETERMINATION OF RELEASE RATE OR TOTAL RELEASE FROM PLANT VENT MONITORS - EP RB-9

5. Determine I-131 Release From Plant Vent Iodine Monitor (RE-24)

a. Determine flow rate ratio

1) Sample flow rate ____________(cfm) (from indicator on sampler)

2) Plant vent flow rate __________(cfm)

3) Ratio (plant vent ÷ sampler = ____________ ÷ ____________ = [ ]

b. Determine curie release of I-131 to the moment

NOTE: Readings must be taken with front panel toggle switch in "$\mu Ci(x10^{-6})$"

1 Chart Reading	2 Calibration Conversion	3 Scale Factor	4 Flow Rate Ratio	5 Iodine Plateout Factor	6 I-131 Release (Ci) 1 x 2 x 3 x 4 x 5
	10^{-12}			1.1*	

c. Projecting total release of I-131

1) Release rate at the moment

1 Chart Reading	2 Calibration Conversion	3 Scale Factor	4 Flow Rate Ratio	5 Iodine Plateout Factor	6 I-131 Release (Ci) 1 x 2 x 3 x 4 x 5
	10^{-17}			1.1*	

2) Projected additional release

$Q_{additional} = (1.44)(Q_{131})(T_{1/2,}cu) = (1.44)(\quad)(\quad) =$ __________ Ci

3) Total release

$Q_{total} = Q_{to\ moment} + Q_{additional} =$ _______ + _______ = [ ] Ci

6. Estimation of I-131 Release by High Alarm Method

a. Background Radiation Levels

Location	Readings (R/hr_{Bkgd})

b. Sample Flow Off

Off t_1	On t_2	Total Time Off $t_1 - t_2$ (min)

*Iodine Plateout Correction Factor of 1.1 Based on 0.3 Micron Particle Size

TITLE: WORK SHEET FOR DETERMINATION OF RELEASE RATE OR TOTAL RELEASE FROM PLANT VENT MONITORS - EP RB-9

c. Total I-131 Curies (Cartridge door closed)

1 R/M	2 R/M_{Bkgd}	3 1 - 2 (R/hr_{net})	4 Distance (Meters)	5 Conversion Factor	6 Iodine Plateout Factor	3 x 4 x 5 x 6 Q(Ci)
				60	1.1*	

d. Total I-131 Curies (Cartridge door open)

1 R/M	2 R/M_{Bkgd}	3 1 - 2 (R/hr_{net})	4 Distance (Meters)	5 Conversion Factor	6 Iodine Plateout Factor	3 x 4 x 5 x 6 Q(Ci)
				60	1.1*	

7. Estimation of I-131 Release Rate by High Alarm Method

1 Q(Ci)	2 Δt (sec)	3 Plant Vent Flow Rate(CFM)	4 Sampler Flowrate	$\frac{1}{2} \times \frac{3}{4}$ Q Ci/sec

*See Page 2

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

TITLE: WORK SHEET FOR RELEASE RATE ESTIMATION FROM CONTAINMENT HIGH RANGE AREA MONITORS - EP RB-9

PERFORMED BY______________________________ DATE__________ TIME______________

PROCEDURE

1. Noble Gas Release Rate

A. Containment high range monitor readings at time = t (hr)

t (hr)	RE-30 Exposure Rate (R/hr)		RE-31 Exposure Rate (R/hr)		Sum of Exposure Rate (R/hr)		① Actual Mean Containment Exposure Rate* (R/hr)
		+		=		x 0.5 =	
		+		=		x 0.5 =	
		+		=		x 0.5 =	
		+		=		x 0.5 =	
		+		=		x 0.5 =	
		+		=		x 0.5 =	

*NOTE: If the reading from one of the two monitors is suspected to be artificially high as a result of proximity to a "hot spot," utilize the reading believed to be most representative of containment ambient radiation levels.

B. Figure 4 exposure rates at time = t (hr)

t (hr)	② DBA-LOCA Containment Exposure Rate at time = t (R/hr)

C. Figure 5 DBA-LOCA Noble Gas Release Rates at time = t (hr)

t (hr)	③ DBA-LOCA Noble Gas Release Rate at time = t (Ci/sec)

TITLE: WORK SHEET FOR RELEASE RATE ESTIMATION FROM CONTAINMENT HIGH RANGE AREA MONITORS - EP RB-9

1. D. Calculation of Noble Gas Release Rate at time = t (hr):

t (hr)	1 Actual Containment Mean Exposure Rate (R/hr)	2 DBA-LOCA Exposure Rate at time = t (R/hr)	1/2 Ratio of Actual/DBA Containment Exposure Rate at time = t		3 DBA-LOCA Noble Gas Release Rate at time = t (Ci/sec)		Noble Gas Projected Accident Release Rate at time = t (Ci/sec)
				x		=	
				x		=	
				x		=	
				x		=	
				x		=	
				x		=	

2. I-131 Release Rate

a. Use mean exposure rates 1 from Noble Gas section, 1.A., if calculated. Otherwise, utilize section 1.A. to perform those steps.

b. Use Figure 4 exposure rates 2 from Noble Gas section, 1.B., if calculated. Otherwise, perform that operation presently.

c. Figure 6 DBA-LOCA I-131 Release Rates at time = t (hr):

t (hr)	4 DBA-LOCA I-131 Release Rate (Ci/sec)

2. D. Calculation of projected I-131 release rates at time = t (hr) after accident:

t (hr)	1/2 Ratio of Actual/DBA Exposure Rates at time = t		4 DBA-LOCA I-131 Release Rate time = t (Ci/sec)		Projected Accident Release Rate at item = t (Ci-sec.)
		x		=	
		x		=	
		x		=	
		x		=	
		x		=	

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

TITLE: WORKSHEET FOR USE OF MAIN STEAM LINE MONITORS, OR RCS COOLANT SAMPLE RESULTS DURING S/G TUBE RUPTURE ACCIDENT

PERFORMED BY______________________ DATE__________ TIME__________

APPLICABILITY

One of the most difficult accidents to estimate the release rate for is a S/G Tube Rupture. This is due to uncertainties in primary to secondary leak rate and in steam release rate to the atmosphere. The release pathway for a secondary system release will largely depend on identifying the affected Steam Generator and whether or not it's associated Main Steam Isolation Valve (MSIV) has been closed. When the MSIV is open and the condenser is used as the principal heat sink, the condenser air ejectors release via the Plant Vent, which is a monitored release point. If the Condenser is unavailable or if the MSIV is closed, steam flow from the affected Steam Generator should be assumed to be released via atmospheric steam dumps. The instantaneous release rate (Q) may be calculated below using the radioactivity concentration of the primary or secondary system.

INSTRUCTIONS

1. Determine the effluent stream concentration, C(μCi/cc) by one of the two following methods:

   1.A. Main Steam Line Monitors (RE-71, 72, 73 & 74). This is the preferred method, but it is possible that radioactivity levels in secondary system steam (or water if the steam-line is flooded) may be too low for Monitors RE-71 - 74 to come on-scale. Otherwise, enter appropriate data below and perform the indicated calculations:

TIME_______ BY_______	Main Steam Line Monitor Channel* RE-71	RE-72	RE-73	RE-74
(1) Reading (CPM)				
(2) Total Conc (C_{Total}) From Figure A**				

*NOTE: Re-71 corresponds to Steam Generator (S/G) number 1, RE-72 to S/G-2, RE-73 to S/G-3, and RE-74 to S/G-4.

**Use emergency (Initial Release Curve) values.

TITLE: WORKSHEET FOR USE OF MAIN STEAM LINE MONITORS, OR RCS COOLANT SAMPLE RESULTS DURING S/G TUBE RUPTURE ACCIDENT

To calculate the fraction of total steam line activity attributable to Noble Gases, Iodines, and Particulates, some assumptions and calculations have to be performed. If the FSAR design basis steam generator tube rupture is assumed, then the fraction of the total activity in the steam line can be broken down as follows:

$C_{NOB\ GAS} = C_{TOTAL} \times 0.9990(F^1_{NG})$

$C_{IODINE} = C_{TOTAL} \times 0.000195(F^1_{I})$ - DEFAULT VALUES

$C_{PART} = C_{TOTAL} \times 0.000005(F^1_{PART})$

However, these fractions should be adjusted based on reactor coolant sample analyses, and partitioning factors. To accomplish this, first determine the relative concentrations of noble gases, iodines, and particulates in the reactor coolant as described below:

Calculate Fractional Reactor Coolant Activity

$F_{NG} = \frac{\text{(Total Noble Gas Activity of Reactor Coolant)}}{\text{(Total Reactor Coolant Activity)}} =$ ____________

$F_{I} = \frac{\text{(Total Iodine Activity of Reactor Coolant)}}{\text{(Total Reactor Coolant Activity)}} =$ ____________

$F_{PART} = \frac{\text{(Total Particulate Activity of Reactor Coolant)}}{\text{(Total Reactor Coolant Activity)}} =$ ____________

Calculate Fractional Steam-Line Activity

$$F^1_{NG} = \frac{(F_{NG})(PF_{NG})}{[(F_{NG})(PF_{NG}) + (F_I)(PF_I) + (F_{PART})(PF_{PART})]}$$

$$F^1_{I} = \frac{(F_{I})(PF_{I})}{[(F_{NG})(PF_{NG}) + (F_I)(PF_I) + (F_{PART})(PF_{PART})]}$$

$$F^1_{PART} = \frac{(F_{PART})(PF_{PART})}{[(F_{NG})(PF_{NG}) = (F_I)(PF_I) + (F_{PART})(PF_{PART})]}$$

TITLE: WORKSHEET FOR USE OF MAIN STEAM LINE MONITORS, OR RCS COOLANT SAMPLE RESULTS DURING S/G TUBE RUPTURE ACCIDENT

VALUES OF STEAM GENERATOR PF (PARTIONING FACTOR)

Steam Generator Water Level

Radionuclide Group	Empty	Normal	Flooded
Noble Gases (PF_{NG})	1.0	1.0	1.0
Iodines (PF_I)	0.1	0.01	1.0
Particulates (PF_{PART})	0.01	0.01	1.0

Calculate Steam-Line Concentration

$C_{NOB\ GAS} = C_{TOTAL} \times F^1_{NG} =$ ____________ $\frac{\mu Ci}{cc}$

$C_{IODINE} = C_{TOTAL} \times F^1_{I} =$ ____________ $\frac{\mu Ci}{cc}$

$C_{PARTICULATES} = C_{TOTAL} \times F^1_{PART} =$ ____________ $\frac{\mu Ci}{cc}$

Proceed to Section 2.

1.B. Steam line activity can be estimated using an analysis of reactor coolant activity. This method is not as preferable as using RE-71, 72, 73, or 74, but it can provide a conservative methodology for calculating steam line concentration.

Calculate Fractional Reactor Coolant Activity

$F_{NG} = \frac{\text{(Total Noble Gas Activity of Reactor Coolant)}}{\text{(Total Reactor Coolant Activity)}} =$ ____________

$F_{I} = \frac{\text{(Total Iodine Activity of Reactor Coolant}}{\text{(Total Reactor Coolant Activity)}} =$ ____________

$F_{PART} = \frac{\text{(Total Particulate Activity of Reactor Coolant)}}{\text{(Total Reactor Coolant Activity)}} =$ ____________

TITLE: WORKSHEET FOR USE OF MAIN STEAM LINE MONITORS, OR RCS COOLANT SAMPLE RESULTS DURING S/G TUBE RUPTURE ACCIDENT

Calculate Fractional Steam-Line Activity

$$F^1_{NG} = \frac{(F_{NG})(PF_{NG})}{[(F_{NG})(PF_{NG}) + (F_I)(PF_I) + (F_{PART})(PF_{PART})]}$$

$$F^1_{I} = \frac{(F_{I})(PF_{I})}{[(F_{NG})(PF_{NG}) + (F_I)(PF_I) + (F_{PART})(PF_{PART})]}$$

$$F^1_{PART} = \frac{(F_{PART})(PF_{PART})}{[(F_{NG})(PF_{NG}) + (F_I)(PF_I) + (F_{PART})(PF_{PART})]}$$

PF Values are the same as provided in section 1A.

Calculate Steam-Line Concentration (Use if Steam Generator is not flooded)

$C_{NOB\ GAS}$ = (Total reactor coolant activity) x F^1_{NG} x 0.056* = ________ $\frac{\mu Ci}{cc}$

$C_{IODINES}$ = (Total reactor coolant activity) x F^1_{I} x 0.056* = ________ $\frac{\mu Ci}{cc}$

$C_{PARTICULATES}$ = (Total reactor coolant activity) x F^1_{PART} x 0.056* = ________ $\frac{\mu Ci}{cc}$

Calculate Steam-Line Concentration (Use if Steam-Generator is flooded)

$C_{NOB\ GAS}$ = (Total reactor coolant activity) x F^1_{NG} = ________ $\frac{\mu Ci}{cc}$

$C_{IODINES}$ = (Total reactor coolant activity) x F^1_{I} = ________ $\frac{\mu Ci}{cc}$

$C_{PARTICULATES}$ = (Total reactor coolant activity) x F^1_{PART} = ________ $\frac{\mu Ci}{cc}$

2. Once the radioactive steam or water concentration is determined in the steam-line using either section 1A or 1B, the release rate from the affected steam generator can be calculated in three different ways. They are described in this section. If the steam generator is flooded (solid) proceed directly to section 2C.

*Factor to account for density differences between steam and water, at approximately 558°F and 1115 (psia). This corresponds to the highest lift setpoint for the safety valves. This value will overestimate the concentration of radioactive material in steam for any lower pressure and temperature.

TITLE: WORKSHEET FOR USE OF MAIN STEAM LINE MONITORS, OR RCS COOLANT SAMPLE RESULTS DURING S/G TUBE RUPTURE ACCIDENT

2A. Steam Flow Rate Monitors

Each steam line at DCPP has a flow rate monitor which is capable of providing steam flow in pounds per hr (lbs/hr) and which will read on-scale if any steam line relief valves are open. Determine this release from the operators for the appropriate channel (512, 522, 532, 542).

FR = ____________ (lbs/hr)

Release Rates for noble gases, iodines, and particulates can then be calculated as:

$$\dot{Q}_{NG} = FR \ \frac{lbs}{hr} \times C_{NOB\ GAS} \ \frac{\mu Ci}{cc} \times 3.1 \times 10^{-6}* = \_\_\_\_\_\_\_\_\_\_ \frac{Ci}{sec}$$

$$\dot{Q}_{IODINE} = FR \ \frac{lbs}{hr} \times C_{IODINE} \ \frac{\mu Ci}{cc} \times 3.1 \times 10^{-6} = \_\_\_\_\_\_\_\_\_\_ \frac{Ci}{sec}$$

$$\dot{Q}_{PARTICULATES} = FR \ \frac{lbs}{hr} \times C_{PARTICULATE} \ \frac{\mu Ci}{cc} \times 3.1 \times 10^{-6} = \_\_\_\_\_\_\_\_\_\_ \frac{Ci}{sec}$$

*Factor to convert lbs/hr to cc/sec and µCi/cc to Ci/sec based on specific volume of steam.

2B. Operation of Relief and/or Safety Valves

Each steam line has six relief valves. Determine which valves are open and total the relief capacity of the open valves.

TITLE: WORKSHEET FOR USE OF MAIN STEAM LINE MONITORS, OR RCS COOLANT SAMPLE RESULTS DURING S/G TUBE RUPTURE ACCIDENT

Valve	Relief Capacity (If Open) in (lbs/hr)
10% Steam Dump	4.0×10^5
#1 Safety	8.5×10^5
#2 Safety	8.5×10^5
#3 Safety	8.5×10^5
#4 Safety	8.5×10^5
#5 Safety	8.5×10^5

Total relief from open valves = FR = ________ (lbs/hr)

$$\dot{Q}_{NG} = FR \ \frac{lbs}{hr} \times C_{NOB\ GAS} \ \frac{\mu Ci}{cc} \times 3.1 \times 10^{-6}* = \_\_\_\_\_\_\_\_ \frac{Ci}{sec}$$

$$\dot{Q}_{IODINE} = FR \ \frac{lbs}{hr} \times C_{IODINE} \ \frac{\mu Ci}{cc} \times 3.1 \times 10^{-6} = \_\_\_\_\_\_\_\_ \frac{Ci}{sec}$$

$$\dot{Q}_{PARTICULATES} = FR \ \frac{lbs}{hr} \times C_{PARTICULATE} \ \frac{\mu Ci}{cc} \times 3.1 \times 10^{-6} = \_\_\_\_\_\_\_\_ \frac{Ci}{sec}$$

*Factor to convert lbs/hr to cc/sec and μCi/sa. Based on the specific volume of steam.

2C. Primary to Secondary Leak Rate

Atmospheric release of steam can be estimated if the primary to secondary leak rate is known. If direct determination of this value is not possible, the default value for a steam generator tube rupture is 600 gpm.

FR = ____________ (gpm)

Calculate Equivalent Steam Flow (Use if Steam Generator is not flooded)

$$FR^1 = FR \times 1.13 \times 10^3* = \_\_\_\_\_\_\_\_ \frac{cc}{sec}$$

*Factor to convert gpm (water) to cc/sec (steam)

TITLE: WORKSHEET FOR USE OF MAIN STEAM LINE MONITORS, OR RCS COOLANT SAMPLE RESULTS DURING S/G TUBE RUPTURE ACCIDENT

Calculate Equivalent Water Flow (Use if Steam Generator is flooded)

$FR^1 = FR \times 63.08^{**} = \underline{\hspace{3cm}} \frac{cc}{sec}$

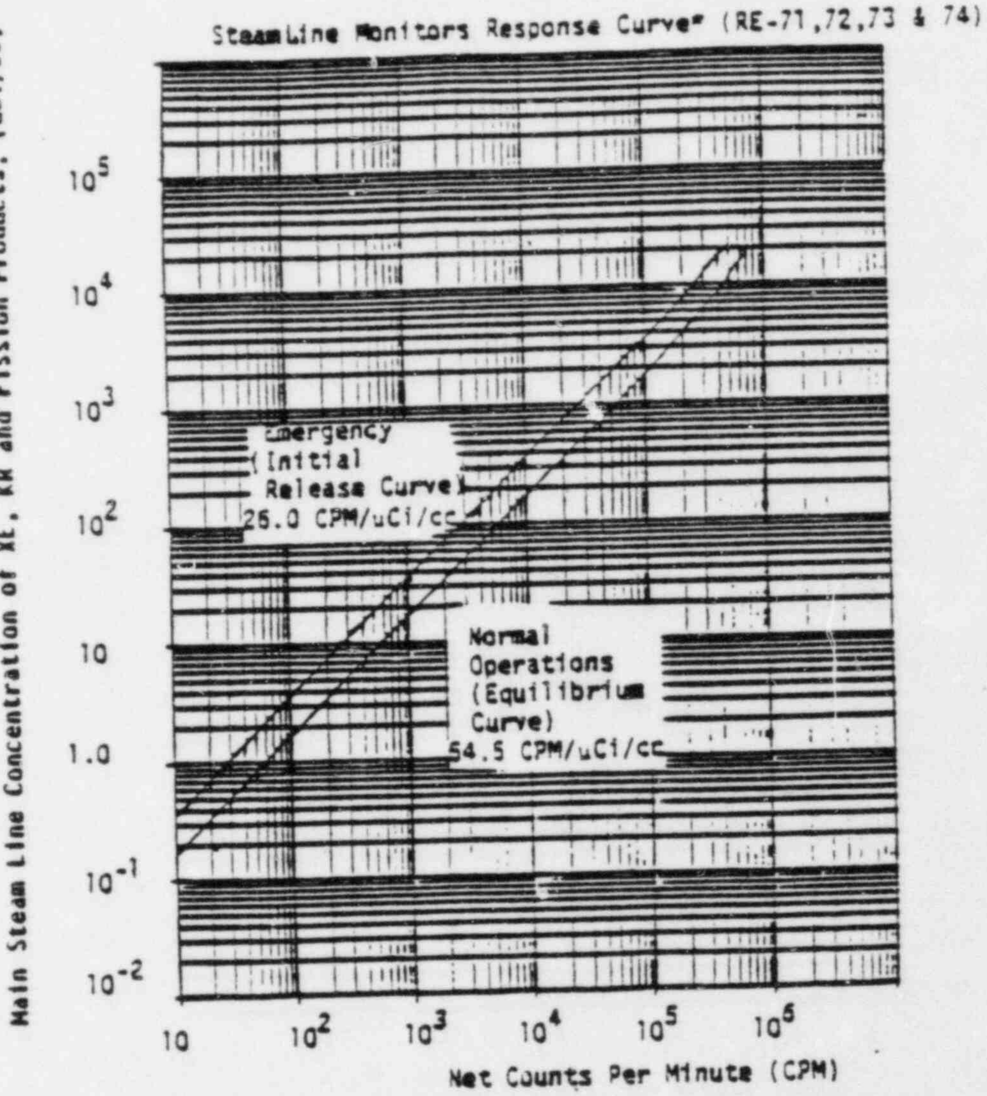
$\dot{Q}_{NG} = FR^1 \frac{cc}{sec} \times C_{NOB\ GAS} \frac{\mu Ci}{cc} \times 10^{-6} = \underline{\hspace{3cm}} \frac{Ci}{sec}$

$\dot{Q}_{IODINE} = FR^1 \frac{cc}{sec} \times C_{IODINE} \frac{\mu Ci}{cc} \times 1 \times 10^{-6} = \underline{\hspace{3cm}} \frac{Ci}{sec}$

$\dot{Q}_{PARTICULATE} = FR^1 \frac{cc}{sec} \times C_{PARTICULATE} \frac{\mu Ci}{cc}\ 1 \times 10^{-6} = \underline{\hspace{3cm}} \frac{Ci}{sec}$

**Factor to convert gpm (water) to cc/sec (water)

FIGURE A



*Isotopic Composition Based On Average Values from FSAR Tables 11.1-11, 12 and 17 for Normal Operations and for Emergency Conditions the Carryover Factors Applied to Normal Iodine Values are 0.13% Volatile Plus 0.25% Mechanical and only 0.25% Mechanical for Particulates.

DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 & 2
EMERGENCY PROCEDURE RB-9
WORK SHEET FOR ESTIMATION OF CURIE RELEASE

CALCULATION BY ______ DATE ______ TIME ______

DESCRIBE ACCIDENT ______ DATE/TIME OF ACCIDENT ______ / ______

A. AVAILABLE ACTIVITY FROM FUEL ASSEMBLY (OR CORE)

1. BASE DATA

ASSM'BY	DATE OF LAST OPERATION	COOLING TIME, t_{cool} (DAYS)	AVG. CORE POWER, Pavg OVER LAST 30 DAYS OF OPER. (FRACTION OF RATED)	NUMBER RUPTURED RODS, N_{rup}	NUMBER OF MELTED RODS N_{melt}

2. AVAILABLE ACTIVITY FROM GAP RELEASE

ISOTOPE	ASSM'BY	① GAP ACT AT FULL POWER (Ci/ROD)	② P_{avg}	③ $\left(t_{cool}/T_{1/2}\right)^1$	④ DECAY FACTOR (FIG.7)[2]	⑤ GAP RELEASE FRACTION	⑥ N_{rup}	⑦ CURIES AVAILABLE ① x ② x ④ x ⑤ x ⑥
Xe-133		34				1.00		
		34				1.00		
		34				1.00		
KR-85		4	1.0	≈0	1.0	1.00		
		4	1.0	≈0	1.0	1.00		
		4	1.0	≈0	1.0	1.00		
I-131		18				0.25		
		18				0.25		
		18				0.25		

3. AVAILABLE ACTIVITY FROM FUEL ROD MELTING

ISOTOPE	ASSM'BY	① ROD ACT AT FULL POWER (Ci/ROD)	② P_{avg}	③ $\left(t_{cool}/T_{1/2}\right)^1$	④ DECAY FACTOR, (FIG.7)[2]	⑤ MELT RELEASE FRACTION	⑥ N_{melt}	⑦ CURIES AVAILABLE ① x ② x ④ x ⑤ x ⑥
XE-133		5000				1.00		
		5000				1.00		
		5000				1.00		
KR-85		22	1.0	≈0	1.0	1.00		
		22	1.0	≈0	1.0	1.00		
		22	1.0	≈0	1.0	1.00		
I-131		2200				0.25		
		2200				0.25		
		2200				0.25		

4. TOTAL AVAILABLE ACTIVITY FROM FUEL DAMAGE

ISOTOPE	GAP RELEASE ASSEMBLY						MELT RELEASE ASSEMBLY						TOTAL AVAILABLE ACTIVITY (CURIES)
XE-133	___	+	___	+	___	+	___	+	___	+	___	=	___
KR-85	___	+	___	+	___	+	___	+	___	+	___	=	___
I-131	___	+	___	+	___	+	___	+	___	+	___	=	___

See notes at end of page 4

B. AVAILABLE ACTIVITY FROM A SAMPLED TANK CONTAINING GAS & LIQUID (WITH MINIMAL VAPORIZATION OF LIQUID)

1. BASE DATA

CONTAINER ________; LIQ.VOL., V_L ________(cc); GAS VOL., V_G ________(cc); ABS. PRESS, P_{con} ________(psia)

LIQ.SAMP. DATE/TIME ________/________; GAS SAMP. DATE/TIME ________/________; DATE/TIME OF RELEASE ________/________

DECAY TIME, t_{dec}, LIQ/GAS ________/________(days)

2. CALCULATION OF AVAILABLE ACTIVITY FROM SAMPLE RESULTS

ISOTOPE	SAMPLE TYPE	① CONC. (μCi/cc)	② $\left(\frac{P_{con}}{1.47}\right)^1$	③ VOLUME, V_L or V_G (cc)	④ INITIAL CONTAINER ACTIVITY(μCi) ①x②x③	⑤ $\left(\frac{t_{dec}}{T_{1/2}}\right)^1$	⑥ DECAY FACTOR (FIG. 7)[2,3]	⑦ CONVERSION (Ci/μCi)	⑧ LIQUID RELEASE FRACTION	⑨ AVAILABLE ACTIVITY (Ci) ④x⑥x⑦x⑧
XE-133	GAS	____	____	____	____	____	____	10^{-6}	1.0	____
	LIQ	____	XXX	____	____	____	____	10^{-6}	1.0	____
KR-85	GAS	____	____	____	____	≈0	1.00	10^{-6}	1.0	____
	LIQ	____	XXX	____	____	≈0	1.00	10^{-6}	1.0	____
I-131	GAS	____	____	____	____	____	____	10^{-6}	1.0	____
	LIQ	____	XXX	____	____	____	____	10^{-6}	10^{-4}	____

C. AVAILABLE ACTIVITY FROM A STEAM RELEASE

1. BASE DATA

CONTAINER ________; PRESSURE ________(psig); SAMPLE DATE/TIME, LIQ ________/________, STM ________/________

BLOWDOWN RATE ________(lb/hr); DURATION ________(hr); TOTAL MASS RELEASE ________(lb)

MASS OF STEAM INITIALLY IN CONTAINER ________(lb), POUNDS OF LIQUID VAPORIZED ________(lb)

2. CALCULATION OF AVAILABLE ACTIVITY FROM SAMPLE RESULTS

ISOTOPE	SAMPLE TYPE	① MASS RELEASE (POUNDS)	② CONC (μCi/gm)	③ CONVERSION (gm/lb)	④ LIQUID RELEASE FRACTION	⑤ CONVERSION (Ci/μCi)	⑥ AVAILABLE ACTIVITY (Ci) ①x②x③x④x⑤
XE-133	STEAM	____	____	454	1.0	10^{-6}	____
	LIQ	____	____	454	1.0	10^{-6}	____
KR-85	STEAM	____	____	454	1.0	10^{-6}	____
	LIQ	____	____	454	1.0	10^{-6}	____
I-131	STEAM	____	____	454	1.0	10^{-6}	____
	LIQ	____	____	454	____	10^{-6}	____

See notes at end of page 4

D. DETERMINATION OF ACTUAL RELEASE FROM AVAILABLE ACTIVITY (Fuel Handling Accident)

① ISOTOPE	② TOTAL AVAILABLE ACTIVITY (Ci)	③ WATER SCRUBBING FACTOR[4,5]	④ FILTER CLEANUP FACTOR[6]	⑤ INITIAL RELEASABLE ACTIVITY AT TIME T_1 ②x③x④ (Ci)
XE-133		1.0	1.0	
KR-85		1.0	1.0	
I-131				

E. DETERMINATION OF EFFECTIVE CLEANUP HALF-LIFE AT TIME, t (Exhaust Ventilation or Containment Sprays Operating)

① ISOTOPE	② RADIOACTIVE[9] DECAY REMOVAL RATE (hrs^{-1})	③ BUILDING[8] FREE VOL. (cc)	④ BUILDING EXHAUST VENT FLOW RATE (cc/sec) WORK SHEET [18-9260]	⑤ BUILDING EXHAUST VENT REMOVAL RATE (sec^{-1}) ④/③	⑥ BUILDING EXHAUST VENT REMOVAL RATE (hr^{-1}) ⑤ x 3600	⑦ CONTAINMENT[7] SPRAY REMOVAL RATE λs (hrs^{-1})	⑧ TOTAL BUILDING REMOVAL RATE (hrs^{-1}) ②x⑥x⑦	⑨ $T_{1/2}$ CU EFFECTIVE CLEANUP HALF-LIFE (HOURS) 0.693/⑧
XE-133	5.45×10^{-3}					0.0		
KR-85	7.38×10^{-6}					0.0		
I-131	3.59×10^{-3}							

F. RELEASABLE ACTIVITY AT A SUBSEQUENT TIME, t_2 FROM RELEASABLE ACTIVITY AT TIME, t_1

① ISOTOPE	② t_1 TIME	③ RELEASABLE ACTIVITY AT TIME, t_1 (Ci)	④ t_2 TIME	⑤ T_E ELAPSED TIME (HOURS) ④-②	⑥ $t_{1/2}$ CU EFFECTIVE CLEANUP HALF-LIFE BETWEEN t_1 and t_2 (HOURS) [Col. 9, E]	⑦ $T_E/T_{1/2}$ CU ELAPSED EFFECTIVE CLEANUP HALF LIVES ⑤/⑥	⑧ DECAY CORRECTION FACTOR [Fig. 7]	⑨ RELEASABLE ACTIVITY AT TIME, T_2 (Ci) ③x⑧
XE-133								
KR-85								
I-131								

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WORK SHEET FOR ESTIMATION OF CURIE RELEASE
EP RB-9

G. BUILDING VENT OR CONTAINMENT LEAKAGE RELEASE RATE AT TIME t_1 or t_2 (Fuel Handling Accident)

① ISOTOPE	② RELEASABLE ACTIVITY AT TIME, T_1, or t_2 (Ci) (Col. 9. F)	③ EXHAUST[10] VENT REMOVAL RATE (sec^{-1}) (Col. 5, E)	④ CONTAINMENT[11] LEAKAGE RATE ($Sec.^{-1}$)	⑤ BUILDING VENT[10] ACTIVITY RELEASE RATE ② x ③ (Ci/sec)	⑥ CONTAINMENT ACTIVITY LEAKAGE RELEASE RATE ② x ④ (Ci/sec)
XE-133			1.2×10^{-8}		
KR-85			1.2×10^{-8}		
I-131			1.2×10^{-8}		

H. NOBLE GAS MIXTURE RELEASE RATES

① ISOTOPE	② EFFECTIVE AGE OF MIXTURE (hr)	③ CALCULATED ISOTOPIC RELEASE RATES (Ci/sec)	④ SUM OF XE-133 AND KR-85 RELEASE RATES (Ci/sec)	⑤ RATIO OF TOTAL NOBLE GAS ACTIVITY TO SUMMED XE-133 AND KR-85 ACTIVITY	④ x ⑤ ESTIMATED NOBLE GAS RELEASE RATE (Ci/sec)
XE-133					
KR-85					
XE-133					
KR-85					
XE-133					
KR-85					

I. NOTES

1. The decay time, t_{dec} or t_{cool} and the half-lives, $T_{1/2}$ are expressed in days.
2. Figure numbers refers to Figures in Emergency Procedure RB-9.
3. Do not consider decay if the container activity has been in equilibrium since the sample was taken.
4. Use 1.0 for I-131 if the leak is above the water level and 0.01 (design value) if the leak is below the water level.
5. The most realistic I-131 value (expected case) is 0.0013. A conservative design basis value is 0.01. In general use design values.
6. The most realistic I-131 value is 99.9% removal, corresponding to a cleanup factor of 0.001. A conservative value is 90% removal, or a factor of 0.01 (design value). In general, use design values.
7. If no spray pumps are functioning, use λ_s = 0.0 hr^{-1}, if one spray pump is functioning, use λ_s = 31.5 hr^{-1}, and If two spray pumps are operating, use λ_s = 92.4 hr^{-1}.
8. Building free volumes - Auxilliary Building = 5.66×10^{10} cc, Fuel Handling Building = 3.71×10^{10} cc, Containment Building = 7.60×10^{10} cc, and Turbine Building = 3.02×10^{11} cc.
9. The radioactive decay removal rates are based on the half-lives of 127 hours for Xe-133, 10.72 yrs. for Kr-85, and 197 hours for I-131.
10. If the accident occurs in containment and containment purge is successfully isolated, it is not appropriate to take credit for exhaust ventilation cleanup.
11. The leakage rate for the containment is based on the design leak rate of 0.1% per day.

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

TITLE - RELEASE RATE SUMMARY-EMERGENCY PROCEDURE RB-9

Calculation by ______________________ Date __________ Time ______________

Describe Release Source ____

____

____

A. Initial Release Rates, Time ____________ am/pm

1. Noble Gas __________ Ci/sec Effective Age ______ (hr)

2. Radioiodine ________ Ci/sec Effective Age ______ (hr)

B. Subsequent Release Rates

1. a. Noble Gas __________ Ci/sec

Time of Release __________ am/pm Effective Age ________ (hr)

b. Noble Gas __________ Ci/sec

Time of Release __________ am/pm Effective Age ________ (hr)

c. Noble Gas __________ Ci/sec

Time of Release __________ am/pm Effective Age ________ (hr)

2. a. Radioiodine ___________ Ci/sec

Time of Release __________ am/pm Effective Age ________ (hr)

b. Radioiodine ___________ Ci/sec

Time of Release __________ am/pm Effective Age ________ (hr)

c. Radioiodine ___________ Ci/sec

Time of Release __________ am/pm Effective Age ________ (hr)

COPY

PACIFIC GAS AND ELECTRIC COMPANY

PGandE 77 BEALE STREET, SAN FRANCISCO, CALIFORNIA 94106 TELEPHONE (415) 781-4211

March 5, 1984

PGandE Letter No.: DCL-84- 090

Mr. John B. Martin, Regional Administrator
U. S. Nuclear Regulatory Commission, Region V
1450 Maria Lane, Suite 210
Walnut Creek, CA 94506-5368

Re: Docket No. 50-275, OL-DPR-76
Docket No. 50-323
Diablo Canyon Units 1 and 2
Emergency Implementing Procedures Updates

Dear Mr. Martin:

In accordance with Section V, "Implementing Procedures," of 10 CFR 50, Appendix E, PGandE is submitting one copy of the updates to the detailed Implementing Procedures for the Diablo Canyon Power Plant Units 1 and 2 Emergency Plan as listed in Attachment 1. Concurrently, two copies of each update are being submitted to the Document Control Desk.

Some of the updates contain privacy/proprietary information. This privacy/proprietary information has been bracketed in accordance with NRC Generic Letter 81-27 and is identified in Attachment 2.

Kindly acknowledge receipt of the above material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

ORIGINAL SIGNED BY

J. O. Schuyler

Enclosures

cc w/enc: R. Fish, NRC (Region V)
Document Control Desk (2)
Service List

cc w/o enc: G. W. Knighton

X005
1/1

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

March 23, 1984

50-275/323 Diablo Canyon

MEMORANDUM FOR: Chief, Document Management Branch, TIDC

FROM: Director, Division of Rules and Records, ADM

SUBJECT: REVIEW OF UTILITY EMERGENCY PLAN DOCUMENTATION

The Division of Rules and Records has reviewed the attached document and has determined that it may now be made publicly available.

J. M. Felton, Director
Division of Rules and Records
Office of Administration

Attachment: As stated