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October 11, 2001

PG&E Letter DCL-01-102

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Emergency Plan Implementing Procedure Update

Dear Commissioners and Staff:

In accordance with Section V, "Implementing Procedures," of 10 CFR 50, Appendix E, enclosed is an update to the emergency plan implementing procedures for Diablo Canyon Power Plant, Units 1 and 2.

As provided under 10 CFR 50.54(q), the changes in this update do not decrease the effectiveness of the emergency plan and, therefore, have been made without prior NRC approval. The plan, as changed, continues to meet the standards of 10 CFR 50.47(b) and 10 CFR 50, Appendix E.

This update contains privacy/proprietary information that has been bracketed in accordance with NRC Generic Letter 81-27.

If there are any questions regarding this update, please contact Mr. Mark Lemke of my staff at (805) 545-4787.

Sincerely,

for James R. Becker

Enclosures

A X45

cc: Ellis W. Merschoff - w/a (2)
David L. Proulx
Girija S. Shukla

DDM/1345

LOCATION OF PRIVACY/PROPRIETARY INFORMATION IN EMERGENCY PLAN IMPLEMENTING PROCEDURES FOR DIABLO CANYON POWER PLANT, UNITS 1 AND 2

Procedure Number	Privacy/ Proprietary Information	Title/Location of Privacy/Proprietary Information
EP G-1 Revision 30	No	Emergency Classification and Emergency Plan Activation
EP G-2 Revision 24	No	Activation and Operation of the Interim Site Emergency Organization (Control Room)
EP G-5 Revision 9	Yes	Evacuation of Nonessential Site Personnel Page 1 of Attachment 7.3
EP RB-14 Revision 6	No	Core Damage Assessment Procedure

DIABLO CANYON POWER PLANT EMERGENCY PLAN IMPLEMENTING PROCEDURES

Table of Contents - Emergency Plan Implementing Procedures
Volume 1A (OM10.ID3 only), Volume 1B (OM10.DC1 only), and Volume 3B

*** Procedure included in this submittal**

TITLE: Emergency Classification and Emergency Plan Activation

1 AND 2

10-2-01
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

TABLE OF CONTENTS

SECTION PAGE

1. SCOPE

1.1 This procedure describes accident classification guidelines and Emergency Plan activation responsibilities.

1.2 This procedure was rewritten; therefore, revision bars are not included.

2. DISCUSSION

2.1 The steps required by this procedure are <u>in addition to</u> the steps required to maintain the plant in, or restore the plant to, a safe condition.

2.2 Events not meeting the minimum classification criteria contained in this procedure should be reviewed for reportability in XI1.ID2, "Regulatory Reporting Requirements and Reporting Process."

3. DEFINITIONS

3.1 Emergency Classification Levels (ECLs)

3.1.1 Notification of Unusual Event (NUE) - characterized by off-normal conditions that:

a. May not in themselves be particularly significant from an emergency preparedness standpoint, but could reasonably indicate a potential degradation of the level of safety of the plant if proper action is not taken or if circumstances beyond the control of the operating staff render the situation more serious from a safety stand point. No releases of radioactive material requiring off-site response or monitoring are expected.

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3.1.2 Alert - events in progress or having occurred, involving an actual or potentially substantial degradation of the plant safety level.

a. Small releases of radioactivity may occur (greater than Technical Specification limits for normal operation, but only a small fraction of the EPA Protective Action Guideline (PAG) exposure levels at the site boundary). It is the lowest level where emergency offsite response may be anticipated.

b. The lowest classification level where off-site emergency response is anticipated.

3.1.3 Site Area Emergency (SAE) - events which are in progress or have occurred involving actual or likely major failures of plant functions needed for protection of the public, but a core meltdown situation is not indicated based on current information.

a. Any releases are not expected to exceed EPA Protective Action Guides except near the site boundary. However, because the possible release is significant, care must be taken in alerting offsite authorities to distinguish whether the release is merely potential, likely, or actually occurring. Response of offsite authorities will be guided initially by this determination.

3.1.4 General Emergency (GE) - event(s) in progress or having occurred which indicate:

a. Imminent substantial core degradation or melting .

b. Potential for containment loss.

c. Radioactive releases can be reasonably expected to exceed EPA PAGs off-site for more than the immediate area.

4. RESPONSIBILITIES

4.1 Interim Site Emergency Coordinator (Interim SEC or ISEC) - Control Room Shift Manager is responsible for initial event classification and emergency plan activation. The ISEC may upgrade the event classification until relieved by either the SEC or RM. In addition, the ISEC may downgrade a NUE to no ECL.

4.2 Site Emergency Coordinator (SEC) - The SEC may upgrade the classification of an event until relieved by the Recovery Manager.

4.3 Recovery Manager (RM) - The RM, once staffed, is responsible for upgrading or downgrading ECLs, and may direct the SEC to change ECLs.

5. INSTRUCTIONS

5.1 The Interim Site Emergency Coordinator shall:

5.1.1 Initially classify and declare the event using ONLY the guidance in Attachment 7.1 of this procedure.

NOTE: Simultaneous EALs that increase the probability of release require escalation of the ECL to one level above the higher EAL.

5.1.2 Formally announce all emergency classification declarations to the Control Room, TSC, or EOF, respectively.

5.2 The ISEC or SEC may:

5.2.1 Upgrade the event to a higher ECL until the Recovery Manager arrives at and assumes responsibility in the EOF. However, the ISEC and SEC shall not downgrade an event classified at the Alert or higher level at any time. The ISEC may downgrade a NUE to no ECL.

5.2.2 Only the Recovery Manager may downgrade an ECL at the Alert or higher level according to the most current controlling EAL.

6. RECORDS

6.1 There are no quality or nonquality records generated by this procedure.

7. ATTACHMENTS

7.1 "Emergency Action Level Classification Chart," 09/27/01

8. REFERENCES

8.1 EP EF-1, "Activation and Operation of the Technical Support Center."

8.2 EP EF-2, "Activation and Operation of the Operational Support Center."

8.3 EP EF-3, "Activation and Operation of the Emergency Operations Facility."

8.4 EP OR-3, "Emergency Recovery."

8.5 EP G-3, "Notification of Offsite Agencies and Emergency Response Personnel."

DIABLO CANYON POWER PLANT
EP G-1
ATTACHMENT 7.1

1 AND 2

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
I. FIRE (All Modes)	1. Fire <u>not</u> under control within 15 minutes of initiating fire fighting efforts <u>AND</u> affecting plant equipment or power supplies in or near the Protected Area(s).	1. Fire <u>not</u> under control within 15 minutes of initiating fire-fighting efforts <u>AND</u> threatening the loss of function of any of the following Safety Related systems required for safe shutdown: - Vital Power Supplies: D/Gs, DFOT, Vital 4kV, 480V, 120VAC, or 125VDC - Primary Systems and Auxiliaries: RCS, CCW, RHR, or Charging and Boration - Heat Sinks: AFW, ASW, 10% Dumps, S/G Safeties, or MSIVs - Control Room, Cable Spreading Rooms, or HSDP.	1. Fire causing the complete loss of function of any one of the following safety related systems required for safe shutdown: - Vital Power Supplies: D/Gs, DFOT, Vital 4kV, 480V, 120VAC, or 125VDC - Primary Systems and Auxiliaries: RCS, CCW, RHR, or Charging and Boration - Heat Sinks: AFW, ASW, 10% Dumps, S/G Safeties, or MSIVs - Control Room, Cable Spreading Rooms, or HSDP.	1. Site Emergency Coordinator judges that a fire could cause common damage to plant systems which is determined to have the potential to release radioactive material in quantities sufficient to cause exposures comparable to General Emergency #4.
II. FUEL DAMAGE OR VESSEL DAMAGE (Modes 1-4)	2. Indication of Fuel Damage as shown by: Confirmed RCS sample shows > 100/E μCi/gm specific activity (Tech Spec 3.4.8) <u>OR</u> Confirmed RCS sample shows dose equivalent I-131 activity > Tech Spec limit for Iodine Spike (Tech Spec Fig. 3.4-1). Category II Continued on next page.	2. Indication of Fuel Damage as shown by: Confirmed RCS sample > 300 μCi/cc of equivalent I-131 specific activity <u>OR</u> equivalent fuel failure is measured by exposure rate from systems carrying reactor coolant per EP RB-14 Category II Continued on next page.	See SAE #14 for Steam Line Break Category II Continued on next page.	2. Degraded core with possible loss of coolable geometry as indicated by: 5 or more thermocouple readings > 1200 deg. F. <u>OR</u> LOCA with no indication of ECCS flow <u>AND</u> indication of fuel damage (See Alert #2) <u>OR</u> LOCA with containment rad levels > values for 100% gap release in EP RB-14. Category II Continued on next page.

NOTE: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
II. FUEL DAMAGE OR VESSEL DAMAGE (Modes 1-4) (Continued)	3. Pressurized Thermal Shock is verified by entry into EOP FR-P.1 AND Left of Limit A curve (EOP F-0).			3. Loss of 2 of 3 Fission Product Barriers: A) Indication of fuel damage (See Alert #2) AND Determination of a Steam Generator Tube Rupture (SGTR) which requires entry into EOP E-3 AND Steam release from ruptured S/G, either used for plant cooldown purposes or due to a steamline break. B) Indication of Fuel Damage (See Alert #2) AND Determination of a SGTR requiring entry into EOP E-3 AND Indication of a steam line break inside containment AND High potential for loss of containment integrity (e.g., loss of function of both Containment Spray trains OR loss of function of one Containment Spray train and four CFCUs).

NOTE: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
II. FUEL DAMAGE OR VESSEL DAMAGE (Modes 1-4) (Continued)	3. Pressurized Thermal Shock is verified by entry into EOP FR-P.1 <u>AND</u> Left of Limit A curve (EOP F-0). (Continued)			C) Indication of Fuel Damage (See Alert #2) <u>AND</u> Determination of a SGTR which requires entry into EOP E-3 <u>AND</u> Indication of a steam line break outside containment with inability to isolate the break. D) Potential fuel damage indicated by incore thermocouples >700 deg. F or RVLIS $<32\%$ <u>AND</u> LOCA as indicated by RCS leakage and SI <u>AND</u> Loss of containment integrity.
III. FUEL HANDLING ACCIDENT (All Modes)		3. Fuel Handling Accident causing a release in Containment or the Fuel Handling Building <u>WITH</u> The potential to exceed the criteria listed in Alert #4 or #5.	2. Fuel Handling Accident causing a release in Containment or the Fuel Handling Building <u>WITH</u> The potential to exceed the criteria listed in SAE #3.	
IV. LOSS OF CONTROL OR RELEASE OF RADIOACTIVE MATERIAL (All Modes)	4. Projected dose rate at the Site Boundary (800 meters) is ≥ 0.057 mRem/hr TEDE <u>OR</u> ≥ 0.170 mRem/hr Thyroid CDE for actual or expected release. Category IV Continued on next page.	4. Projected dose rate at the Site Boundary (800 meters) is ≥ 0.57 mRem/hr TEDE <u>OR</u> ≥ 1.7 mRem/hr Thyroid CDE for actual or expected release. Category IV Continued on next page.	3. Projected dose at the Site Boundary (800 meters) is ≥ 100 mRem TEDE <u>OR</u> ≥ 500 mRem Thyroid CDE for actual or expected release. Category IV Continued on next page.	4. Projected dose at the Site Boundary (800 meters) is ≥ 1,000 mRem TEDE <u>OR</u> ≥ 5,000 mRem Thyroid CDE for actual or expected release. Category IV Continued on next page.

<u>NOTE</u>: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
IV. LOSS OF CONTROL OR RELEASE OF RADIOACTIVE MATERIAL (All Modes) (Continued)	5. A valid reading in excess of the isolation setpoint, which fails to isolate the release on any of the Radiological Process Effluent Monitors: RE-18 OR RE-23 During discharge <u>only</u>.	5. Valid alarm on plant vent high range noble gas monitor RE-29. **<u>NOTE</u>**: ALARMS AT STATE OES SACRAMENTO.		
	6. An actual liquid release which exceeds the limits of 10 CFR 20, Appendix B, Table 2, Col. 2 per CY2.ID1.	6. An actual liquid release which exceeds 10x the limits of 10 CFR 20, Appendix B, Table 2, Col. 2 per CY2.ID1.		
	7. Radiological Effluent Process Monitor High Radiation Alarm with valid reading in excess of alarm setpoint on any of the following monitors: RE-14/14R RE-24/24R RE-28/28R.	7. Unplanned or unanticipated increase of 1 R/hr or greater in any of the following areas: Passageways, <u>OR</u> Normally occupied areas, <u>OR</u> Accessible areas normally < 100 mR/hr, <u>OR</u> Outside boundaries of Radiologically Controlled Areas <u>AND</u>, for any area above, a potential exists for <u>EITHER</u> an uncontrolled release to the environment <u>OR</u> a loss of ability to maintain plant safety functions.		
	8. Unplanned or uncontrolled release to the environment exceeding alarm setpoints on RE-3.	8. Unexplained increase of 50 X DAC in airborne radioactivity outside the boundary of the Radiologically Controlled Areas, but within the Plant Protected Area.		

<u>NOTE</u>: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
V. LOSS OF CONTROL ROOM (All Modes)		9. Entry into OP AP-8A, "Control Room Accessibility," AND controls established within 15 minutes.	4. Entry into OP AP-8A, "Control Room Accessibility," AND controls not established within 15 minutes.	
VI. LOSS OF ENGINEERED SAFETY FEATURE	9. Plant is not brought to required operating Mode within any applicable Tech Spec Action Statement time limit (Modes 1-4).		5. Complete loss for greater than 15 minutes of any of the following functions needed to reach or maintain Hot Shutdown (while in Modes 1-4): AFW capability Steam Dump System and S/G Safety Valves Loss of the capability to maintain RCS inventory as evidenced by a loss of all charging pumps coincident with the inability to depressurize and inject with the Safety Injection pumps Loss of capability to increase the Boric Acid concentration sufficient to maintain Keff less than .99 in Mode 4 with a loss of capability to trip control rods ASW or CCW Systems Loss of electrical power or I&C for any of the above listed systems, causing a complete loss of function.	5. Loss of Heat Sink indicated by: Entry into EOP FR H.1 AND Loss of water inventory in 3 S/Gs (< 23% [34%] Wide Range).
	10. Loss of function of both RHR trains for greater than 15 minutes while in Mode 5 or 6.	10. Loss of function of both RHR trains for greater than 15 minutes in Modes 1-4.		
	11. A loss of function of all charging pumps for greater than 15 minutes when normally used for RCS inventory control (Modes 1-4).	11. An unplanned shutdown of the RHR System (while in Mode 5 or 6) for > 1 hour with no other normal means of decay heat removal available (e.g., flooded reactor cavity or steam generators with loops filled).		
		12. An unplanned loss of function of the RHR System (Mode 5 or 6) for greater than 15 minutes AND RCS thermocouple temperature is projected to exceed 200 deg.F within 1 hour of RHR loss (see Appendix B of OP AP SD series) OR RCS thermocouple temperature exceeds 200 deg.F.		

NOTE: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
VII. LOSS OF POWER OR ALARMS OR ASSESSMENT OR COMMUNICATIONS	12. Loss of all off-site power for greater than 15 minutes AND at least 2 D/Gs are supplying their vital busses (Modes 1-4).	13. Loss of all off-site power for greater than 15 minutes AND only 1 D/G is supplying its vital bus (Modes 1-4).	6. Loss of all on-site AND off-site AC power for > 15 minutes (Modes 1-4).	See General Emergency Condition #5 under LOSS OF ENGINEERED SAFETY FEATURE.
	13. Loss of all off-site power for greater than 15 minutes AND at least 1 D/G is supplying its vital bus (Modes 5 and 6).	14. Loss of all off-site and on-site AC power for greater than 15 minutes in Modes 5 or 6.		
	14. Loss of all vital DC power as indicated by DC Bus 11(21), 12(22), and 13(23) undervoltage for > 15 minutes (Modes 5-and 6)	15. Loss of all vital DC power as indicated by DC Bus 11(21), 12 (22) and 13 (23) undervoltage for < 15 minutes (Modes 1-4).	7. Loss of all vital DC power as indicated by DC Bus 11 (21), 12 (22) and 13 (23) undervoltage for > 15 minutes (Modes 1-4).	
	15. Loss of assessment capabilities as indicated by a total loss of SPDS in the Control Room AND simultaneous loss of all displays for any "Accident Monitoring" variable in Tech Spec Table 3.3-10 for > 1 hour while in Modes 1, 2 or 3. *			
	16. Main Control Room Annunciators PKs 1 through 5 AND display capabilities AND typewriter all do not respond to an alarm condition in Modes 1-4 for over 15 minutes..	16. Main Control Room Annunciators PKs 1 through 5 AND display capabilities AND typewriter all do not respond to an alarm condition in MODES 1-4 for over 15 minutes AND the plant is in a significant transient (plant trip, SI, or generator runback >25 Mw/min), nonannunciating systems available.	8. Main Control Room Annunciators PKs 1 through 5 AND display capabilities AND typewriter all do not respond to an alarm condition in MODES 1-4 for over 15 minutes AND the plant is in a significant transient AND backup, nonannunciating systems are not available (PPC, SPDS).	

NOTE: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

* IF THE AFFECTED VARIABLE IN TABLE 3.3-10 IS THE PLANT VENT RADIATION MONITOR-HIGH RANGE, REFER TO THE COMPENSATORY MEASURES IN STP G-16.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
VII. LOSS OF POWER OR ALARMS OR ASSESSMENT OR COMMUNICATIONS (Continued)	17. Total loss of communication capability with off-site agencies (all Modes) as indicated by the inability to communicate with SLO County (by telephone and radio) <u>OR</u> the NRC Operations Center.			
VII. NATURAL PHENOMENA (All Modes)	18. Ground motion felt and recognized as an earthquake by a consensus of Control Room operators on duty <u>AND</u> measuring greater than 0.01g on the Earthquake Force Monitor.	17. Earthquake > 0.2 g verified by Seismic Monitors.	9. Earthquake > 0.4 g verified by Seismic Monitors.	6. Site Emergency Coordinator's judgment that major internal or external events (e.g., earthquakes, wind damage, explosions, etc.) which could cause massive common damage to plant systems which is determined to have the potential to release radioactive material in quantities sufficient to cause exposures comparable to General Emergency #4.
	19. Flooding of any plant structure that causes initiation of entry to Mode 3 due to a Tech Spec action statement.	18. High water exceeding Intake Structure main deck elevation or low water causing cavitation and shutdown of both ASW pumps for < 15 minutes.	10. High water causing flooding of ASW pump compartments or low water causing the shutdown of both ASW pumps for > 15 minutes.	
	20. Tsunami or Hurricane Warning from the State, NOAA, NWS, Coast Guard or System Dispatcher <u>OR</u> Observation of low or high water levels at the Intake Structure indicative of a Tsunami or Hurricane.	19. Sustained wind of 85 mph (38 m/sec) at any elevation on the Met. Tower.	11. Sustained wind speed > 100 mph (45 m/sec).	
	21. A tornado sighted within Site Boundary.	20. Tornado strikes the plant protected area.		

<u>NOTE</u>: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
IX. OTHER HAZARDS (All Modes)	22. Report of airplane crash within the Site Boundary or unusual airplane activity threatening the plant.	21. Confirmed missile, airplane crash or explosion involving a plant structure in the protected area.	12. Missile, airplane crash or explosion causing complete loss of a safety system function that causes entry into a Tech Spec Action Statement.	See General Emergency #6 above.
	23. Confirmed explosion on-site.			
	24. Turbine failure causing casing penetration OR damage to turbine or generator seals	22. Turbine failure generating missiles that cause visual damage to other safety related structures, equipment, controls OR power supplies.		
	25. Significant release of flammable OR toxic gas OR liquid that prevents, even with SCBAs, operations inside the power block OR intake structure (ref. CP M-9a).	23. Release of flammable OR toxic gas OR liquid that jeopardizes operation of safety related systems by either preventing required access OR by threatening imminent damage.		
X. PRIMARY OR PRI/SEC OR SECONDARY LEAK) (Modes 1-4	26. RCS unidentified OR pressure boundary leakage that exceeds 10 gpm OR identified leakage that exceeds 25 gpm.	24. Primary leak rate > 50 gpm.	13. Known primary system LOCA during which RCS subcooling cannot be maintained >20°F OR PZR level cannot be maintained >4% (28% with adverse containment).	See General Emergency #3 under Fuel or Vessel Damage.
	27. SI Actuation with ECCS injection into the RCS resulting from a valid signal based on actual plant conditions. **NOTE**: SI ACTUATION ALSO ALARMS AT OES IN SACRAMENTO.	25. Determination of a SGTR which results in entry into EOP E-3.	14. Determination of a SGTR coincident with steam release from ruptured S/G, either used for plant cooldown purposes or due to a steamline break.	
	28. Steam line break which results in SI actuation.	26. Determination of a steam line break with > 10 gpm Primary to Secondary leakage.		
	29. Failure of a PZR PORV AND Block Valve OR Safety Valve fails to reseat, excluding allowable leakage, following a pressure reduction below the reset point.			

NOTE: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

TITLE: Emergency Action Level Classification Chart

	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
XI. REACTOR PROTECTION SYSTEM FAILURE (Modes 1-4)		27. Anticipated Transient Without Scram (ATWS) as indicated by: Failure of an automatic reactor trip to trip the reactor.	15. An ATWS condition with no fuel damage evident <u>AND</u> An additional failure of a system required for Hot Shutdown (See SAE #5) to actuate.	7. ATWS with Fuel Damage indications (see Alert Condition #2 under FUEL DAMAGE) <u>OR</u> ATWS with potential Core Melt indicated by incore thermocouples > 700 deg. F <u>AND</u> RVLIS < 32%.
XII. SECURITY THREAT (Modes 1-4)	30. Report by Security that a forcible entry or sabotage attempt has occurred or a credible bomb threat has been received against the areas within the protected area boundary.	28. Security reports ongoing security threat involving physical attack on the facility or a sabotage device has been detected that threatens the operability of safety related equipment (see Alert #1).	16. Security reports ongoing physical attack on the facility or a sabotage device causing a confirmed loss of a safety system function that causes an entry into a Tech Spec Action Statement (see SAE #1).	8. Security reports ongoing security threat which causes loss of control of the operations of the plant to hostile forces.
XIII. SITE EMERGENCY COORDINATOR'S JUDGMENT (All Modes)	31. Site Emergency Coordinator determines conditions warrant increased awareness on the part of off-site authorities of initiation of a plant shutdown per Tech Spec LCOs or involve other than normal controlled shutdown.	29. Site Emergency Coordinator judges plant conditions exist that warrant precautionary activation of the TSC and placing the EOF and other key emergency personnel on stand-by.	17. Site Emergency Coordinator judges that conditions exist that warrant activation of the emergency centers and monitoring teams or a precautionary notification to the public near the site.	9. Site Emergency Coordinator judges conditions exist which have a potential to release radioactive material in quantities sufficient to cause exposures comparable to General Emergency #4.

<u>NOTE</u>: SIMULTANEOUS EALS THAT INCREASE THE PROBABILITY OF RELEASE REQUIRE ESCALATION OF THE CLASSIFICATION TO ONE LEVEL ABOVE THE HIGHER EAL.

Activation and Operation of the Interim Site Emergency Organization

9·11·01
Effective Date

Procedure Classification: Quality Related

Table Of Contents

ATTACHMENTS:
1. Notification of Unusual Event Checklist, 05/08/01
2. Termination of Notification of Unusual Event Checklist, 02/09/00
3. Alert Checklist, 12/19/00
4. Site Area Emergency Checklist, 12/19/00
5. General Emergency Checklist, 12/19/00
6. Emergency Evaluation Coordinator Checklist, 12/07/00
7. ISEC/SEC/RM Turnover Checklist, 09/14/00
8. Control Room Assistant Checklist, 08/13/01

1. SCOPE

This procedure provides Emergency Planning actions to be taken by the Control Room during a declared emergency.

This procedure does not direct logging or initialing of actions. OP1.DC37, "Plant Logs" provides Operations logging instructions.

2. DISCUSSION

The checklists are intended to provide quick reference to all possible emergency response actions and require judgment in prioritizing activities based upon available resources and unforeseen circumstances.

3. RESPONSIBILITIES

Shift Manager

Assumes the role of Interim Site Emergency Coordinator (ISEC) The ISEC takes command and control of the emergency response effort until relieved. The ISEC has the responsibility and authority to:

- Declare emergency classifications.
- Notify off-site authorities of the event and make Protective Action Recommendations.
- Conduct assembly and accountability on-site.
- Authorize extraordinary emergency measures such as authorizing emergency response personnel to exceed normal established dose limits.
- Provide direction for all emergency response operations.
- Maintain liaison with off-site authorities.
- Authorize the evacuation of the plant site.
- Approve press releases.
- Initiate on-site and off-site radiological monitoring.

Shift Technical Advisor (STA)

Assumes the role of Emergency Evaluation Coordinator (EEC).
This position performs technical evaluations of plant response, dose assessments, and Protective Action Recommendations (PARs) for approval by the ISEC. The EEC may also issue KI thyroid block as directed by the ISEC.

Shift Foreman or Licensed Operator

Assumes the roles of Emergency Liaison Coordinator (ELC).
This position performs emergency notifications to San Luis Obispo County, California State Office of Emergency Services (OES) and the Nuclear Regulatory Commission (NRC) in accordance with EP G-3 until relieved.

Shift Foreman or SCO

Assumes the role of Interim Emergency Operations Coordinator (IEOC).
This position manages Control Room operational activities and advises the ISEC needed event reclassifications.

Control Room Assistant (CRA)

Perform emergency response personnel call-out, PG&E management emergency notifications and notification of NRC-Resident Inspectors in accordance with EP G-3.

4. INSTRUCTIONS

4.1 INTERIM SITE EMERGENCY COORDINATOR (ISEC)

1) Depending on the emergency classification level, use one of the following ISEC-Checklists:

   - Attachment 1 - Notification of Unusual Event
   - Attachment 2 - Termination of Notification of Unusual Event
   - Attachment 3 - Alert
   - Attachment 4 - Site Area Emergency
   - Attachment 5 - General Emergency

<u>NOTE</u>: If the emergency classification is upgraded, exit the existing attachment and start a new attachment at the higher classification level. For classification downgrades, continue using the checklist from the previous, higher classification level.

2) Use Attachment 7 - ISEC/SEC/RM Turnover Checklist when relieved of ISEC responsibilities.

4.2 EMERGENCY EVALUATION COORDINATOR (EEC)

Use Attachment 6 - Emergency Evaluation Coordinator Checklist until relieved.

<u>NOTE</u>: STA responsibilities for providing advisory technical support take precedence unless directed otherwise by the Interim Site Emergency Coordinator (ISEC).

4.3 INTERIM EMERGENCY OPERATIONS COORDINATOR (IEOC)

1) Manage the Control Room operational activities.

2) Advise the ISEC of needed changes to emergency classifications.

3) Provide a turnover briefing to the Emergency Operations Coordinator upon arrival in the Control Room.

5. RECORDS

Documents generated by this procedure are nonquality good business records and are maintained by the EP group for a period of three years.

Notification of Unusual Event Checklist

______________________________ ______________
SIGNATURE DATE

1) ☐ Approve Emergency Classification.

2) ☐ Announce emergency to Control Room staff and direct staff to assume their emergency response roles.

3) ☐ Direct CRA to initiate VANS (if required). This should be accomplished within 10 minutes after the event has been classified.

4) ☐ Complete ***DCPP Event Notification Form*** (EP G-3, Attachment 9.3).

5) ☐ Complete ***PAR Form*** (EP RB-10).

6) ☐ Direct the Emergency Liaison Coordinator to complete EP G-3, Attachment 9.4 ***Off-Site Notification Log Sheet***.

7) ☐ Make PA Announcement. →

> **Attention all personnel:**
> **The Diablo Canyon Emergency Plan has been activated at the ___________ level for Unit _____, due to ________________________.**
>
> **No actions by emergency response personnel are required at this time.**

8) ☐ If a release of airborne radioactive materials may be occurring or has occurred, direct the Emergency Evaluation Coordinator to perform an off-site dose assessment per EP R-2.

9) ☐ <u>WHEN</u> plant conditions no longer meet any emergency classification criteria, <u>THEN</u> terminate the emergency using the TERMINATION Checklist.

10) ☐ Route this completed checklist to the Emergency Planning Supervisor for retention.

Termination of Notification of Unusual Event Checklist

<u>WHEN</u> plant conditions no longer meet any emergency classification criteria,
<u>THEN</u> terminate the emergency as follows:

1) ☐ Complete DCPP Event Notification Form (EP G-3, Attachment 9.3) for Termination of the NUE.

2) ☐ Direct the Emergency Liaison Coordinator to complete a Termination Notification to all parties originally notified of the NUE (EP G-3, Attachment 9.4 Off Site Notification Log Sheet).

3) ☐ Announce the Termination of the NUE to Control Room staff.

4) ☐ Make PA Announcement. ➡

> **Attention all personnel:**
>
> **The Notification of Unusual Event Emergency Classification at Diablo Canyon has been terminated.**

5) ☐ Continue to support any Casualty Procedures in effect, as the Shift Manager, but without the authority of the ISEC.

6) ☐ Ensure an "AT REPT" Action Request is written within 24 hours of the termination of the event.

7) ☐ Route this completed checklist to the Emergency Planning Supervisor for retention.

______________________________ SIGNATURE

______________ DATE

Alert Checklist

SIGNATURE ______________________ DATE ____________

1) ☐ Approve Emergency Classification.

2) ☐ Announce emergency to Control Room staff and, unless previously done, direct staff to assume their emergency response roles.

3) ☐ Direct CRA to initiate VANS. This should be accomplished within 10 minutes after the event has been classified.

4) ☐ Complete **DCPP Event Notification Form** (EP G-3, Attachment 9.3).

5) ☐ Complete **PAR Form** (EP RB-10).

Note: If KI needs to be issued prior to the TSC being activated, then refer to EP RB-3.

6) ☐ Direct the Emergency Liaison Coordinator to make Off-Site Agency Notifications. (EP G-3, Attachment 9.4 **Off-Site Notification Log Sheet**.)

7) ☐ Make PA Announcement. →

8) ☐ Sound the Site Emergency Signal for 60 sec.

9) ☐ Repeat the PA announcement. →

> **Attention all personnel:**
> **The Diablo Canyon Emergency Plan has been activated (or upgraded) at the ___________ level for Unit _____, due to**
> **_____________________.**
> **Emergency Plan position holders report immediately to your assigned response location.**

10) ☐ Contact DCPP Watch Commander (if required) to:

- Implement Personnel assembly, accountability and Site Access control during emergencies (EP G-4).
- Initiate Evacuation of non-essential personnel (EP G-5).

11) ☐ If a release of airborne radioactive materials may be occurring or has occurred, direct the Emergency Evaluation Coordinator to perform an off-site dose assessment per EP R-2.

12) ☐ Ensure emergency responsibilities are turned over to the Emergency Response Organization as soon as possible. Use **ISEC / SEC / RM Turnover Checklist**.

13) ☐ Announce to the Control Room that Emergency Response Facilities have been activated.

14) ☐ Route this completed checklist to the Emergency Planning Supervisor for retention.

Site Area Emergency Checklist

SIGNATURE ______________________ DATE __________

1) ☐ Approve Emergency Classification

2) ☐ Announce emergency to Control Room staff and, unless previously done, direct staff to assume their emergency response roles.

3) ☐ Direct CRA to initiate VANS. This should be accomplished within 10 minutes after the event has been classified.

4) ☐ Complete **DCPP Event Notification Form** (EP G-3, Attachment 9.3).

5) ☐ Complete **PAR Form** (EP RB-10).

Note: If KI needs to be issued prior to the TSC being activated, then refer to EP RB-3.

6) ☐ Direct the Emergency Liaison Coordinator to make Off-Site Agency Notifications. (EP G-3, Attachment 9.4 **Off-Site Notification Log Sheet.**)

7) ☐ Make PA Announcement. →

8) ☐ Unless already sounded, sound the Site Emergency Signal for 60 sec.

9) ☐ Repeat the PA announcement. →

Attention all personnel:
The Diablo Canyon Emergency Plan has been activated (or upgraded) at the __________ level for Unit _____, due to ______________________.
Emergency Plan position holders report immediately to your assigned response location.

10) ☐ Contact DCPP Watch Commander (if required) to:

- Implement **Personnel assembly, accountability and Site Access control during emergencies** (EP G-4).
- Initiate **Evacuation of non-essential personnel** (EP G-5).

11) ☐ If a release of airborne radioactive materials may be occurring or has occurred, direct the Emergency Evaluation Coordinator to perform an off-site dose assessment per EP R-2.

12) ☐ Ensure emergency responsibilities are turned over to the Emergency Response Organization as soon as possible. Use **ISEC / SEC / RM Turnover Checklist**.

13) ☐ Announce to the Control Room that Emergency Response Facilities have been activated.

14) ☐ Route this completed checklist to the Emergency Planning Supervisor for retention.

General Emergency Checklist

________________________________ ____________
SIGNATURE **DATE**

1) ☐ Approve Emergency Classification.

2) ☐ Announce emergency to Control Room staff and, unless previously done, direct staff to assume their emergency response roles.

3) ☐ Direct CRA to initiate VANS. This should be accomplished within 10 minutes after the event has been classified.

4) ☐ Complete **DCPP Event Notification Form** (EP G-3, Attachment 9.3).

5) ☐ Complete **PAR Form** (EP RB-10).

Note: If KI needs to be issued prior to the TSC being activated, then refer to EP RB-3.

6) ☐ Direct the Emergency Liaison Coordinator to make Off-Site Agency Notifications. (EP G-3, Attachment 9.4 **Off-Site Notification Log Sheet**.)

7) ☐ Make PA Announcement. →

8) ☐ Unless already sounded, sound the Site Emergency Signal for 60 sec.

9) ☐ Repeat the PA announcement. →

> **Attention all personnel:**
> **The Diablo Canyon Emergency Plan has been activated (or upgraded) at the __________ level for Unit _____, due to**
> **____________________________.**
> **Emergency Plan position holders report immediately to your assigned response location.**

10) ☐ Contact DCPP Watch Commander (if required) to:

- Implement **Personnel assembly, accountability and Site Access control during emergencies** (EP G-4).
- Initiate **Evacuation of non-essential personnel** (EP G-5).

11) ☐ If a release of airborne radioactive materials may be occurring or has occurred, direct the Emergency Evaluation Coordinator to perform an off-site dose assessment per EP R-2.

12) ☐ Ensure emergency responsibilities are turned over to the Emergency Response Organization as soon as possible. Use **ISEC / SEC / RM Turnover Checklist**.

13) ☐ Announce to the Control Room that Emergency Response Facilities have been activated.

14) ☐ Route this completed checklist to the Emergency Planning Supervisor for retention.

Emergency Evaluation Coordinator Checklist

_______________________________ SIGNATURE __________ DATE

INITIAL ACTIONS

1) ☐ IF fulfilling the STA function and CSFTs have been entered, THEN continue to provide advisory technical support as required by Emergency Operating Procedures.

2) ☐ IF a radiological release is indicated, THEN notify the ISEC.

CONTINUING ACTIONS

1) ☐ IF directed by the ISEC, THEN implement EP R-2 to perform an assessment of Site Boundary Dose and Dose Rate.
   a) Provide the completed calculations to the ISEC for review.
   b) Compare the dose and dose rate calculation results with the EALs in EP G-1 and recommend changes in classification to the ISEC.

2) ☐ Implement EP RB-10 and provide a completed PAR Form 69-13216 to the ISEC for review and approval.

NOTE: Steps 1 and 2 shall be repeated at least every 30 minutes or immediately if release status changes significantly until directed otherwise by the ISEC.

3) ☐ IF the Interim SEC directs the deployment of on-site and off-site monitoring teams, THEN follow the guidelines in EP RB-8.

4) ☐ IF the Interim SEC directs the administration of KI, THEN issue KI per EP RB-3, performing the notification, supply and documentation tasks of the Radiological Advisor and Liaison Coordinator in that procedure.

5) ☐ Activate ERDS on SPDS within 30 minutes of emergency declaration.

TURNOVER OF EEC DUTIES (ALERT OR HIGHER ONLY)

1) ☐ When contacted by the EOF Radiological Manager, provide a briefing of plant status and reactor core conditions.

2) ☐ When contacted by the EOF Radiological Manager, provide a briefing of radiological conditions and the status of field monitoring teams, and, as applicable, the status of KI issue.

3) ☐ WHEN notified by the Shift Manager that SEC functions have been transferred, THEN exit this procedure.

ISEC/SEC/RM Turnover Checklist

______________________________ SIGNATURE

__________ DATE

☐ Time of Event: ______________ ☐ Time VANS Activated: ______________

☐ Time Facility *must* be Activated: ______________

☐ Current Classification: ______________

☐ Reason for Classification: (See Notification form)

☐ Affected Unit: ☐ Unit 1 ☐ Unit 2

☐ Unit 1 Status Mode: __________ Power: __________

☐ Unit 2 Status Mode: __________ Power: __________

☐ Fission barriers challenged: ☐ Fuel ☐ RCS ☐ Containment

☐ Release in progress: ☐ Yes ☐ No

☐ Release source:

Last Notification number: __________	Last PAR number: __________
Time it was completed: __________	Time it was completed: __________
Next Notification number: __________	Next PAR number: __________
Time it is required: __________	Time it is required: __________
Who will do it? ☐ CR ☐ TSC ☐ EOF	Who will do it? ☐ CR ☐ TSC ☐ EOF

Site accountability status	☐ Complete	☐ In Progress	Due: __________
Early work release initiated	☐ Yes	☐ No	Status: __________
Plant evacuation initiated	☐ Yes	☐ No	Status: __________

NOTES

Control Room Assistant Checklist

______________________________ ____________
SIGNATURE **DATE**

When instructed or given a copy of the DCPP Event Notification Form (69-10581) by the ISEC or Shift Manager complete the following steps:

Event Notification Time = T____________

1) ☐ Activate VANS as soon as possible but <u>within 10 minutes</u> of the <u>initial</u> event notification ONLY. (Verify success when your pager is activated).

   (T + 10) = ________________

2) ☐ Call the County and State <u>within 15 minutes</u> of the classification time written on the DCPP Event Notification Form and document on the Off-Site Notifications Log Sheet.

   (T + 15) = ________________

3) ☐ At the sounding of the Site Emergency Signal, perform Control Room Assembly and Accountability per EP G-4 <u>within 30 minutes.</u> Fax to DCPP Watch Commander at 3115.

   (Site Signal + 30) = ________________

4) ☐ Fax copies of the DCPP Event Notification Form and the Protective Action Recommendations Form to the County, State, NRC, and EPIM (549-9187).

<u>NOTE</u>: When talking with the County or State, do not volunteer information or explanations about the emergency beyond what is on the approved notification form.

5) ☐ Fax copies of the DCPP Event Notification Form and the Protective Action Recommendations Form to the EOF and TSC.

PACIFIC GAS AND ELECTRIC COMPANY — NUMBER EP RB-14
NUCLEAR POWER GENERATION — REVISION 6
DIABLO CANYON POWER PLANT —
EMERGENCY PLAN IMPLEMENTING PROCEDURE — UNITS 1 AND 2

TITLE: Core Damage Assessment Procedure

9-28-01
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

1. SCOPE

1.1 This procedure is used to estimate the extent of clad and/or core failure following an emergency situation involving inadequate core cooling. Westinghouse Owner's Group "Post Accident Core Damage Assessment Methodology" was used as a reference for preparing this procedure.

2. DISCUSSION

2.1 Fuel damage resulting in the release of radioactive material can occur following a loss of coolant accident (LOCA) or loss of available heat sinks. These events, if uncorrected, can lead to localized or widespread overheating of the fuel and eventual clad and/or core failure.

2.2 This procedure provides an initial detection of potential core damage and a preliminary and a long-term methodology for assessing core damage.

2.2.1 The initial detection of core damage can be done by measuring the radiation level at a distance of one foot from the center of the letdown line in the letdown heat exchanger room as shown in Attachment 8.3. Should the radiation level exceed 15 R/hr then fuel damage is indicated at the Alert #2 emergency action level.

2.2.2 The preliminary assessment uses parameters such as reactor vessel water level and core temperatures to confirm that conditions exist which can lead to clad and/or core failure. This is quantified through the use of containment hydrogen and area radiation monitor readings.

2.2.3 Long-term methodology uses reactor coolant and containment air sample analysis to determine the extent of clad and/or core failure more accurately. Long-term sampling will require about 30 days.

3. RESPONSIBILITIES

3.1 The Emergency Radiological Advisor (ERA) is responsible for the implementation of this procedure. The preliminary assessment can be initiated while awaiting sample analysis results necessary for the long-term assessment. Refer to Table 1 for recommended sample locations.

4. PRELIMINARY ASSESSMENT (FORM 69-10422)

4.1 GENERAL INFORMATION

Record the information requested.

4.2 INADEQUATE CORE COOLING

4.2.1 Indication of Conditions - Check the appropriate response to the questions.

4.2.2 Evaluation of Conditions - The more boxes checked the greater potential for inadequate core cooling. Proceed to Step 4.3 and continue monitoring the situation.

4.3 CONTAINMENT RADIATION LEVELS

If loss of Reactor Coolant is not occurring skip this and the next section and proceed to Step 5, LONG TERM ASSESSMENT.

4.3.1 Containment Area Radiation Monitors Operable

a. Record containment area radiation monitor readings, R/hr, in the spaces labeled RE-30 READING and RE-31 READING.

b. Multiply the RE-30 and RE-31 monitor readings. Record the results $(R/Hr)^2$ in the space labeled READINGS PRODUCT.

c. Take the square root of the READINGS PRODUCT and record the result (R/Hr) in the space labeled AVERAGE READING. If only one monitor is operable, use that monitor's reading as the AVERAGE READING.

4.3.2 Containment Area Radiation Monitors Inoperable

a. Obtain exposure rate, R/hr, outside the equipment hatch or personnel hatch (outside airlock). Use a portable ionization chamber.

b. Record the exposure rate into the space labeled EQUIPMENT or PERSONNEL HATCH READING.

4.3.3 Percent Clad and/or Core Failure Estimate

Containment Area Radiation Monitors Operable

a. Obtain the 100% Gap and Core Release exposure rates, R/hr, from Figures 2 and 3. Record these values into the spaces labeled 100% GAP RELEASE AND 100% CORE RELEASE. Use Step 4.1.2 time after reactor shutdown.

b. To determine the percent clad failure multiply the AVERAGE READING by 100 and divide the result by the 100% GAP RELEASE. Record the result (%) in the space labeled PERCENT CLAD FAILURE.

c. To determine the percent core failure multiply the AVERAGE READING by 100 and divide the result by the 100% CORE RELEASE. Record the result (%) in the space labeled PERCENT CORE FAILURE.

4.3.4 Containment Area Radiation Monitors Inoperable

a. Perform Step a. above, recording the values in spaces F and H.

b. Obtain the 100% Gap and Core Release exposure rates, R/hr, from Figures 4 or 5 for the location selected in Step 4.3.2. Record these values into the spaces labeled 100% GAP RELEASE and 100% CORE RELEASE. Use Step 4.1.2 time after reactor shutdown.

c. To determine the percent clad failure multiply the EQUIPMENT HATCH or PERSONNEL HATCH READING from Step 4.3.2, by 100 and divide the result by the 100% GAP RELEASE. Record the result (%) in the space labeled PERCENT CLAD FAILURE (G and K).

d. To determine the percent core failure multiply the EQUIPMENT or PERSONNEL HATCH READING from Step 4.3.2, by 100 and divide the result by the 100% CORE RELEASE. Record the result (%) in the space labeled PERCENT CORE FAILURE (I and M).

e. If the PERCENT CLAD FAILURE is ≤100%, multiply F x G and enter the result in space N. If the PERCENT CLAD FAILURE is >100%, multiply H x I and enter the result in space N.

4.4 CONTAINMENT HYDROGEN LEVELS

If core failure was indicated in Step 4.3, proceed to Step 5.0, LONG TERM ASSESSMENT.

4.4.1 Percent Clad Failure Estimate

NOTE 1: Hydrogen levels in containment are only a valid indicator of damage within the first 24 hours of the accident, assuming that the hydrogen recombiners are not operating. If assessment from the area radiation monitors and the H_2 monitors differ, utilize data from Reactor Vessel Level Indication System (RVLIS), etc., to select the most representative assessment. If resolution cannot be obtained, use the highest estimated level of percent clad failure.

NOTE 2: Immediately notify the Emergency Evaluation and Recovery Coordinator of any increase in hydrogen reading. An increase requires the starting of the installed hydrogen recombiners in containment. If the hydrogen monitor reading exceeds 3%, evaluate using the Hydrogen Purge System and obtaining external recombiners. This action should be taken because internal recombiners may become an ignition source if hydrogen concentrations exceeds 4.0%.

a. Record containment hydrogen monitor readings, %, in the spaces labeled CEL-82 READING and CEL-83 READING.

b. Get the monitor readings average and record the result in the space labeled AVERAGE READING. If only one monitor is operable, use that monitor's reading as the AVERAGE READING.

5. LONG TERM ASSESSMENT (FORM 69-10423)

5.1 GENERAL INFORMATION

Record the information requested.

5.2 EMERGENCY CORE COOLING SYSTEM (ECCS) VOLUME INJECTED

5.2.1 Reactor Coolant System (RCS) Sample

a. Use Table 1 to select the RCS sampling location (hot leg or cavity sump) corresponding to Step 4.1.2 postulated accident.

b. Request an isotopic analysis of the selected RCS sample.

5.2.2 Density Correction Factor

a. Record the RCS TEMPERATURE, °F, in the space provided.

b. Use Figure 7 and the RCS TEMPERATURE to determine the DENSITY CORRECTION FACTOR. Record this value in the space provided.

5.2.3 Dilution Volume

a. Determine the Refueling Water Storage Tank (RWST) volume, gal, prior to the accident. Record this value in the space labeled PRIOR RWST VOLUME.

b. Obtain the current RWST volume, gal. Record this value in the space labeled CURRENT RWST VOLUME.

c. To determine the RWST volume injected, cc, subtract the CURRENT RWST VOLUME from the PRIOR RWST VOLUME and multiply the result by the CONVERSION FACTOR (3,785 cc/gal). Record this value in the space labeled INJECTED RWST VOLUME.

d. Determine the number of accumulators discharged. Record this number in the space labeled ACCUMULATOR QUANTITY.

e. To determine the accumulator volume injected multiply the number of accumulators discharged by the accumulator volume (4.28E7cc). Record the result, cc, in the space labeled INJECTED ACCUMULATOR VOLUME.

f. To determine the dilution volume, cc, sum the INJECTED RWST VOLUME, INJECTED ACCUMULATOR VOLUME, and the REACTOR COOLANT SYSTEM VOLUME (3.56E8 cc). Record the result in the space labeled DILUTION VOLUME.

5.3 LIQUID INVENTORY

a. Obtain the selected RCS sample analysis results (Step 5.2.1) for the specified isotopes (off gases must be included). Record the sample activity concentration, uCi/cc, in the spaces labeled SAMPLE ACTIVITY CONCENTRATION.

b. Record the DENSITY CORRECTION FACTOR (Step 5.2.2) and the DILUTION VOLUME (Step 5.2.3) into the spaces provided.

c. To determine the liquid inventory, Ci, multiply the SAMPLE ACTIVITY CONCENTRATION, DENSITY CORRECTION FACTOR, DILUTION VOLUME, and the CONVERSION FACTOR (1E-6 Ci/uCi). Record the result in the space labeled LIQUID INVENTORY.

5.4 PRESSURE AND TEMPERATURE CORRECTION FACTOR

5.4.1 Containment Atmosphere Sample

a. Use Table 1 to select the containment atmosphere sampling location corresponding to Step 4.1.2 postulated accident.

b. Request an isotopic analysis of the selected sample.

5.4.2 Pressure and Temperature Correction Factor

a. Record the containment atmosphere pressure, psig, and temperature,°R, in the spaces labeled CONTAINMENT ATMOSPHERE PRESSURE and CONTAINMENT ATMOSPHERE TEMPERATURE (psia = psig + 14.7, and °R = °F + 460).

b. Record the containment atmosphere sample pressure, psia, and temperature, °R, in the spaces labeled SAMPLE PRESSURE and SAMPLE TEMPERATURE (psia = psig + 14.7, and °R = °F + 460).

c. Divide the CONTAINMENT ATMOSPHERE PRESSURE by the SAMPLE PRESSURE. Record the result in the space labeled PRESSURE RATIO.

d. Divide the SAMPLE TEMPERATURE by the CONTAINMENT ATMOSPHERE TEMPERATURE. Record the result in the space labeled TEMPERATURE RATIO.

e. To determine the pressure and temperature correction factor multiply the PRESSURE RATIO and the TEMPERATURE RATIO. Record the result in the space labeled PRESSURE TEMPERATURE CORRECTION FACTOR.

5.5 GASEOUS INVENTORY

a. Obtain the selected containment atmospheric sample analysis result (Step 5.4.1) for the specified isotopes. Record the activity concentration, uCi/cc, in the spaces labeled SAMPLE ACTIVITY CONCENTRATION.

b. Record the PRESSURE TEMPERATURE CORRECTION FACTOR (Step 5.4.2) into the space provided.

c. To determine the gaseous inventory, Ci, multiply the SAMPLE ACTIVITY CONCENTRATION, PRESSURE TEMPERATURE CORRECTION FACTOR, CONTAINMENT VOLUME (7.36E10 cc), and the CONVERSION FACTOR (1E-6 Ci/uCi). Record the result in the space labeled GASEOUS INVENTORY.

5.6 INVENTORY CORRECTION - CONSTANT POWER HISTORY

This section is to be used when the power level has remained relatively constant (within ± 10 percent) for 30 days prior to reactor shutdown. If the power level has not been relatively constant, proceed to Section 5.7.

5.6.1 Power Level

a. Determine the average power level (%) for the 4 days prior to reactor shutdown. Record this value into the space labeled POWER LEVEL_1.

b. Determine the average power level (%) for the 30 days prior to reactor shutdown. Record this value into the space labeled POWER LEVEL_2.

c. Determine the operation time (days) since the previous reactor shutdown. Record this value into the space labeled OPERATION TIME.

d. Determine the average power level (%) for the operation time. Record this value into the space labeled AVERAGE POWER LEVEL.

5.6.2 Power Correction Factor

a. Divide the POWER LEVEL_1 (Step 5.6.1) by 100 and record the result in the space labeled POWER CORRECTION FACTOR_1.

b. Divide the POWER LEVEL_2 (Step 5.6.1) by 100 and record the result in the space labeled POWER CORRECTION FACTOR_2.

c. Obtain the Cs-134 power correction factor from Figure 8. Use Step 5.6.1 operation time and average power level. Record the obtained value into the space labeled Cs-134 POWER CORRECTION FACTOR.

5.6.3 Corrected Inventory - Gap

a. To determine the gap corrected inventory multiply the GAP EQUILIBRIUM INVENTORY by the applicable POWER CORRECTION FACTOR (FACTOR_1, FACTOR_2, OR Cs-134).

b. Record the results, Ci, into the space labeled GAP CORRECTED INVENTORY.

5.6.4 Corrected Inventory - Core

a. To determine the core corrected inventory multiply the CORE EQUILIBRIUM INVENTORY by the applicable POWER CORRECTION FACTOR ($FACTOR_1$, $FACTOR_2$, or Cs-134).

b. Record the results, Ci, into the spaces labeled CORE CORRECTED INVENTORY.

5.7 INVENTORY CORRECTION FOR VARIABLE POWER HISTORY

When the power level has not remained relatively constant for 30 days prior to reactor shutdown, the effects of power changes must be taken into account.

5.7.1 Power Level

a. Determine the operation time (days) since the previous reactor shutdown. Record this value into the space labeled OPERATION TIME.

b. Determine the average power level (%) for the operation time. Record this value into the space labeled AVERAGE POWER LEVEL.

5.7.2 Power Correction Factor

a. Obtain the Cs-134 power correction factor from Figure 8. Use Step 5.7.1 operation time and average power level. Record the obtained value into the space labeled Cs-134 POWER CORRECTION FACTOR.

b. Use the following equation to determine the power correction factor for other isotopes.

$$PF_i = \frac{\sum_j P_j\left(1 - e^{-\lambda_i t_{1j}}\right)e^{-\lambda_i t_{2j}}}{RP} \qquad \text{Eq. I}$$

Where:

PF_i = 30 day Power Correction Factor for nuclide i,

P_j = average power level (Mwt) for time t_{1j},

RP = rated power level of the core (Mwt),

RP = 3411 Mwt Unit 1

RP = 3411 Mwt Unit 2

λ_i = decay constant (days^{-1}) nuclide i,

t_{1j} = time (days) where power does not vary more than ±10%, and

t_{2j} = time (days) from end of t_{1j} to 30-days.

5.7.3 Gap Corrected Inventory

a. To determine the gap corrected inventory multiply the GAP EQUILIBRIUM INVENTORY by the applicable POWER CORRECTION FACTOR (Step 5.7.2).

b. Record the results, Ci, into the spaces labeled GAP CORRECTED INVENTORY.

5.7.4 Core Corrected Inventory

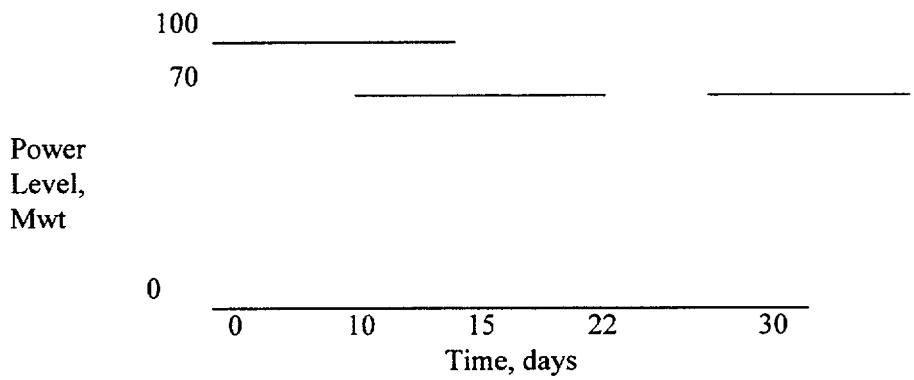
a. To determine the core corrected inventory multiply the CORE EQUILIBRIUM INVENTORY by the applicable POWER CORRECTION FACTOR (Step 5.7.2).

b. Record the results, Ci, into the spaces labeled CORE CORRECTED INVENTORY.

5.7.5 Power Correction Factor Sample Calculation:

Unit 1 has operated for the 30 days prior to reactor shutdown with the following power history:

100	Mwt	for	10	days
70	Mwt	for	5	days
0	Mwt	for	7	days
70	Mwt	for	8	days



Unit 1 rated power level = 3411 Mwt

Period 1

P_1 = 100 Mwt, t_{11} = 10 days, t_{21} = 20 days

Period 2

P_2 = 70 Mwt, t_{12} = 5 days, t_{22} = 15 days

Period 3

P_3 = 0.0 Mwt, t_{13} = 7 days, t_{23} = 8 days

Period 4

P_4 = 70 Mwt, t_{14} = 8 days, t_{24} = 0 days

The equation for Nuclide i would be:

$$PF_i = ([100\,(1-e^{-\lambda_i\,(10)})e^{-\lambda_i\,(20)}]$$
$$+\ [\,70\,(1-e^{-\lambda_i\,(5)})e^{-\lambda_i\,(15)}]$$
$$+\ [0.0\,(1-e^{-\lambda_i\,(7)})e^{-\lambda_i\,(8)}]$$
$$+\ [\,70\,(1-e^{-\lambda_i\,(8)})e^{-\lambda_i\,(0)}]) \div 3411$$

5.8 RELEASE PERCENT, IODINE RATIO, AND NOBLE GAS RATIO

5.8.1 Gap Released Percent

a. Obtain the LIQUID INVENTORY, Ci, from step 5.3 and record into the spaces provided.

b. Obtain the GASEOUS INVENTORY, Ci, from step 5.5 and record into the spaces provided.

c. Add the LIQUID INVENTORY and the GASEOUS INVENTORY. Record the result into the space labeled RELEASED INVENTORY.

d. Obtain the CORRECTED INVENTORY from step 5.6.3 or 5.7.3 and record into the spaces provided.

e. To determine the release percent divide the RELEASED INVENTORY by the CORRECTED INVENTORY and multiply the result by 100. Record this value into the spaces labeled RELEASE PERCENT.

5.8.2 Core Release Percent

a. Obtain the LIQUID INVENTORY, Ci, from step 5.3 and record into the spaces provided.

b. Obtain the GASEOUS INVENTORY, Ci, from step 5.5 and record into the spaces provided.

c. Add the LIQUID INVENTORY and the GASEOUS INVENTORY. Record the result into the space labeled RELEASED INVENTORY.

d. Obtain the CORRECTED INVENTORY from step 5.6.4 or 5.7.4 and record into the spaces provided.

e. To determine the release percent divide the RELEASED INVENTORY by the CORRECTED INVENTORY and multiply the result by 100. Record this value into the spaces labeled RELEASE PERCENT.

5.8.3 Iodine Ratio

a. Obtain the I-133 and I-131 RELEASED INVENTORY from step 5.8.1. Record these values, Ci, into the spaces provided.

b. To determine the iodine ratio divide the I-133 RELEASED INVENTORY by the I-131 RELEASED INVENTORY. Record the result into the space labeled IODINE RATIO.

5.8.4 Noble Gas Ratio

a. Obtain the Kr-87 and Xe-133 RELEASED INVENTORY from step 5.8.1. Record these values, Ci, into the spaces provided.

b. To determine the noble gas ratio divide the Kr-87 RELEASED INVENTORY by the Xe-133 RELEASED INVENTORY. Record the result into the space labeled NOBLE GAS RATIO.

5.9 ASSESSMENT WORKSHEET

5.9.1 Assessment Worksheet

a. Obtain the RELEASE PERCENT (step 5.8.1and 5.8.2) for the specified isotopes and mark the applicable Assessment Worksheet box.

b. Obtain the IODINE and NOBLE GAS RATIOS (Step 5.8.3 and 5.8.4) and mark the applicable Assessment Worksheet box.

c. Determine damage category base on mark distribution on the Assessment Worksheet.

6. REFERENCES

6.1 Westinghouse Owners Post Accident Core Damage Assessment Methodology.

6.2 PG&E Calculation PAM-0-07-065, Rev. 2, 2/22/97 "Core Exit Temperature Indication Uncertainty".

6.3 PG&E Calculation PAM-0-07-403, "RCS Wide Range Pressure Indication Uncertainty".

7. APPENDICES

7.1 Table 1 Recommended Sample Locations

7.2 Figure 1 Saturation Curve

7.3 Figure 2 Post LOCA Exposure Rate Inside Containment From Noble Gases (RE-30 and RE-31) - Gap Release

7.4 Figure 3 Post LOCA Exposure Rate Inside Containment From Noble Gases (RE-30 and RE-31) - Core Release

7.5 Figure 4 Post LOCA Exposure Rate Outside of Equipment Hatch From Noble Gases - Gap and Core Release

7.6 Figure 5 Post LOCA Exposure Rate Outside of Personnel Hatch From Noble Gases - Gap and Core Release

7.7 Figure 6 Percent H_2 in Containment vs Percent Clad Failure

7.8 Figure 7 Density Correction Factor

7.9 Figure 8 Power Correction Factor for Cs-134

8. ATTACHMENTS

8.1 Form 69-10422, "Preliminary Assessment," 09/19/01

8.2 "Long Term Assessment," 09/19/01

8.3 "Initial Detection of Potential Core Damage," 09/19/01

9. SPONSOR

Alex Taylor

APPENDIX 7.1

TABLE 1

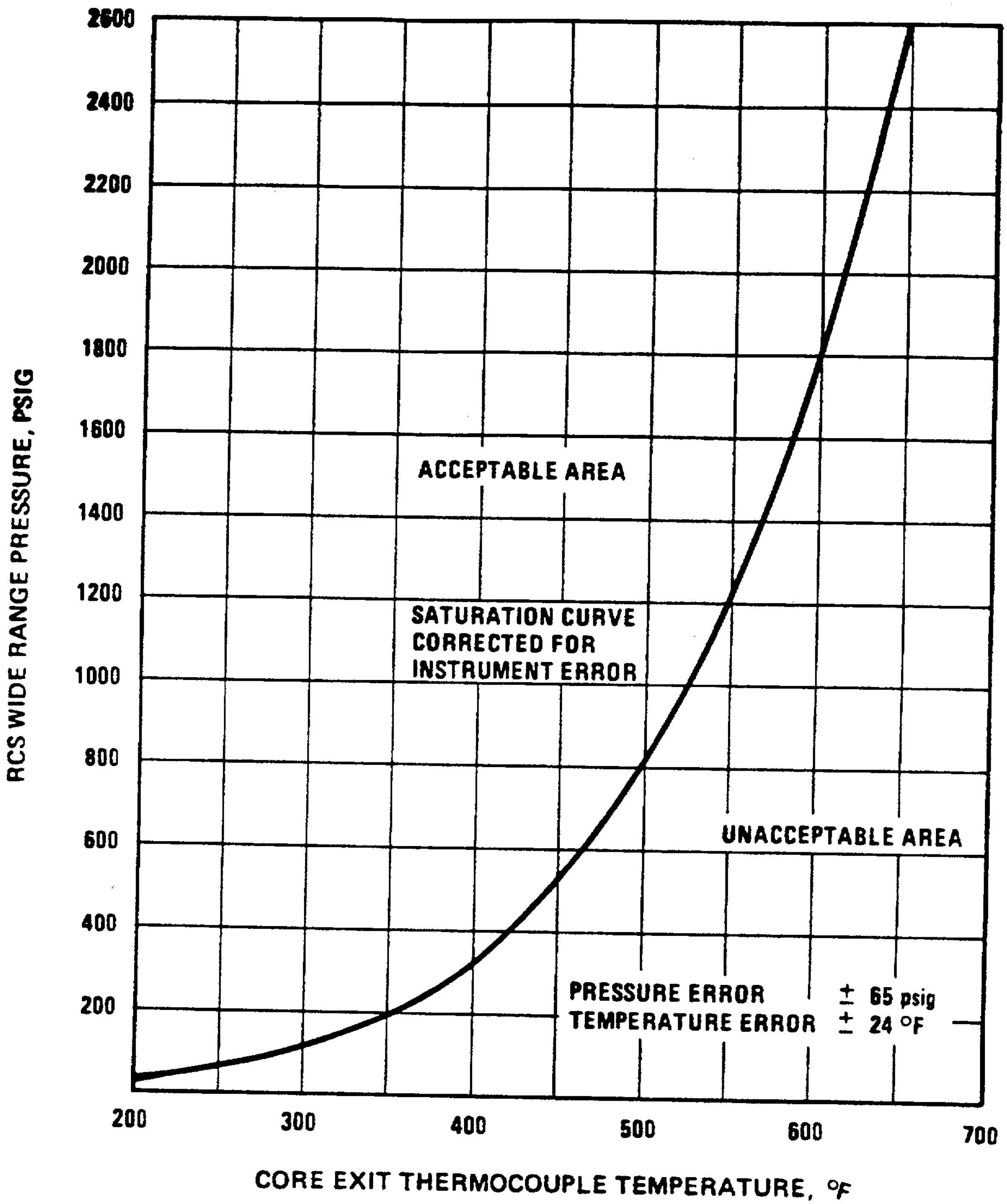
Recommended Sample Locations

Postulated Accident	Sampling Locations
- Small break LOCA	
Reactor Power > 1%	RC Hot Log 1 or 4 (LSP) Containment air (CASP)
Reactor Power < 1%	RC Hot Leg 1 or 4 (LSP)
- Large break LOCA	
Reactor Power > 1%	Reactor Cavity Sump (LSP) Containment air (CASP) RC Hot Leg 1 or 4
Reactor Power < 1%	Reactor Cavity Sump (LSP) Containment air (CASP)
- Steam Line break	RC Hot Leg 1 or 4 (LSP) Containment air (CASP)
- Steam Generator tube rupture	RC Hot Leg 1 or 4 (LSP) Containment air (CASP)
- Indication of significant Containment Sump Inventory	Reactor Cavity Sump (LSP) Containment air (CASP)
- Containment building Radiation Monitor Alarm	Reactor Cavity Sump (LSP) Containment air (CASP)
- Safety injection actuated	RC Hot Leg 1 or 4 (LSP)
- Indication of High Radiation Level in RCS	RC Hot Leg 1 or 4 (LSP)

APPENDIX 7.2

FIGURE 1

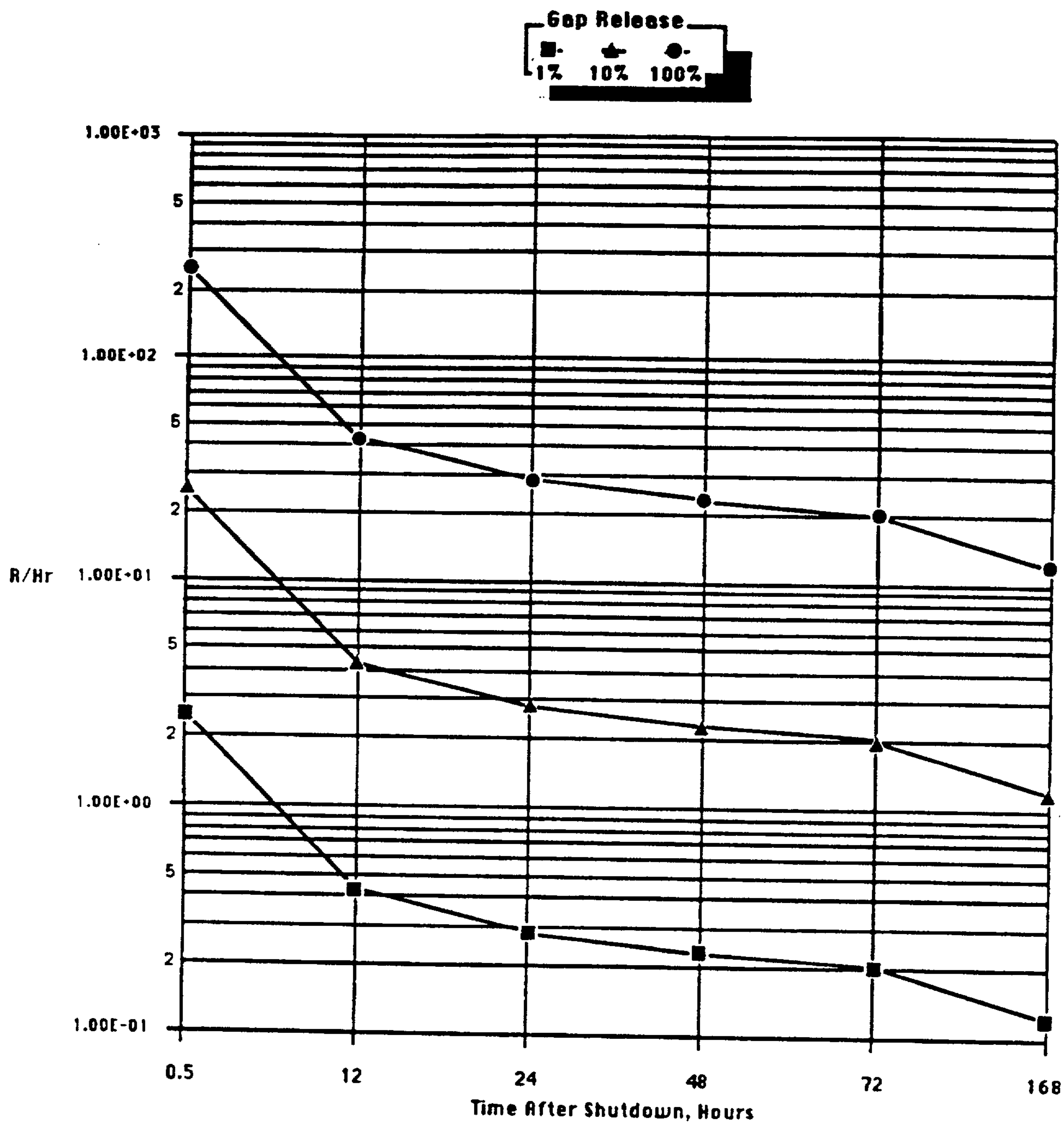
SATURATION CURVE



APPENDIX 7.3

FIGURE 2

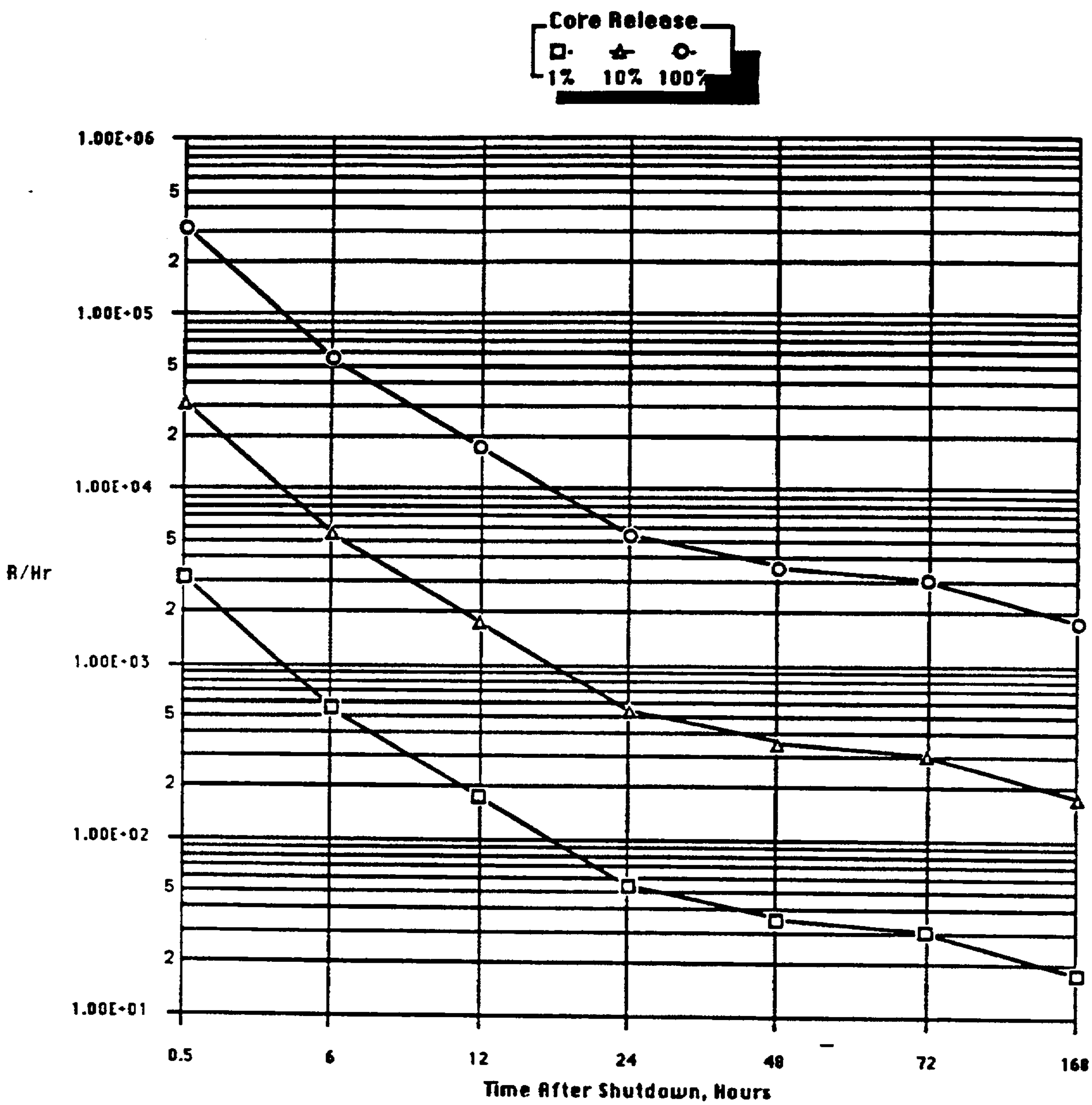
POST LOCA EXPOSURE RATE INSIDE CONTAINMENT FROM NOBLE GASES
(RE-30 AND RE-31) - GAP RELEASE



APPENDIX 7.4

FIGURE 3

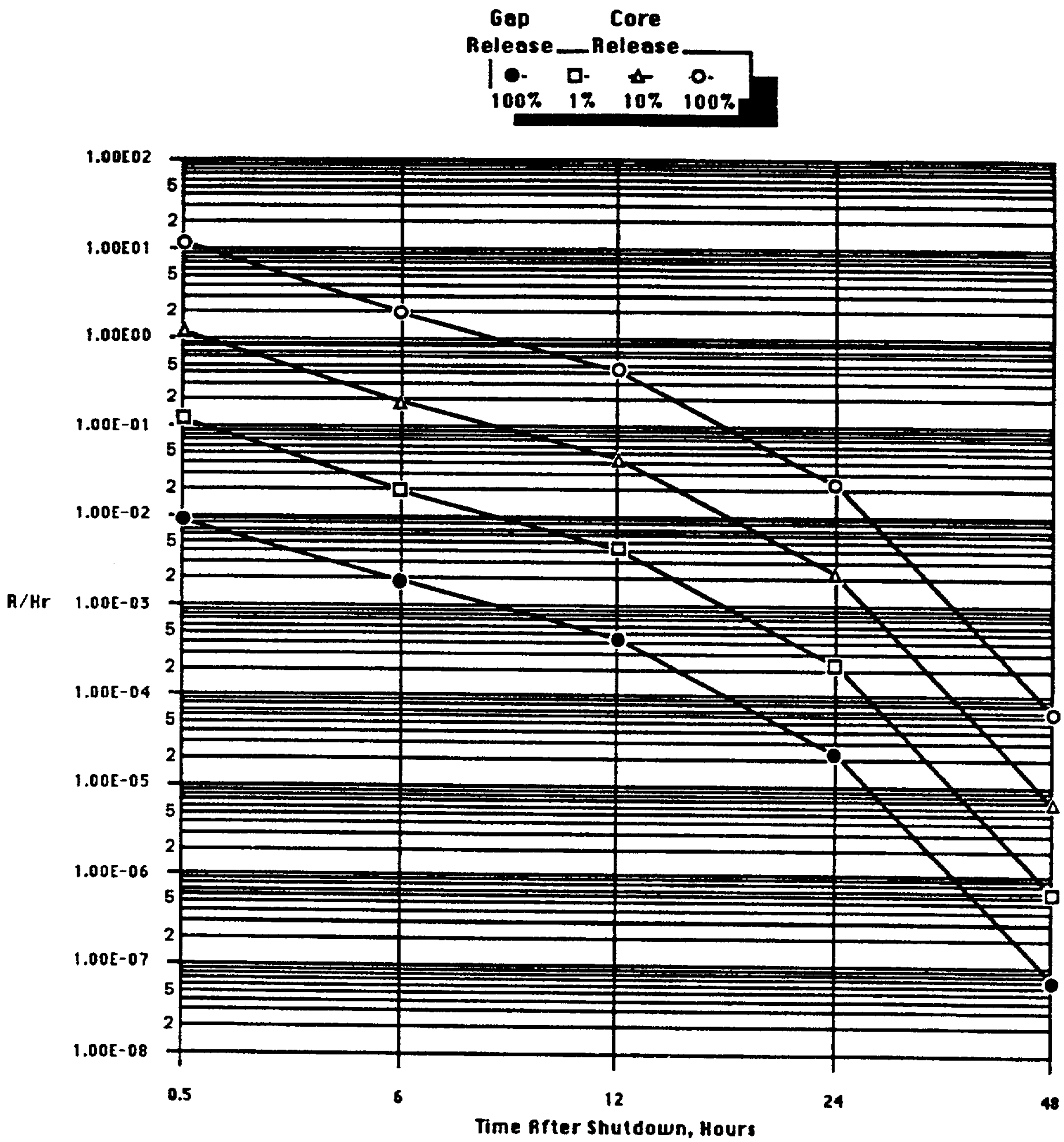
POST LOCA EXPOSURE RATE INSIDE CONTAINMENT FROM NOBLE GASES (RE-30 AND RE-31) - CORE RELEASE



APPENDIX 7.5

FIGURE 4

POST LOCA EXPOSURE RATE OUTSIDE OF EQUIPMENT HATCH FROM NOBLE GASES - GAP AND CORE RELEASE

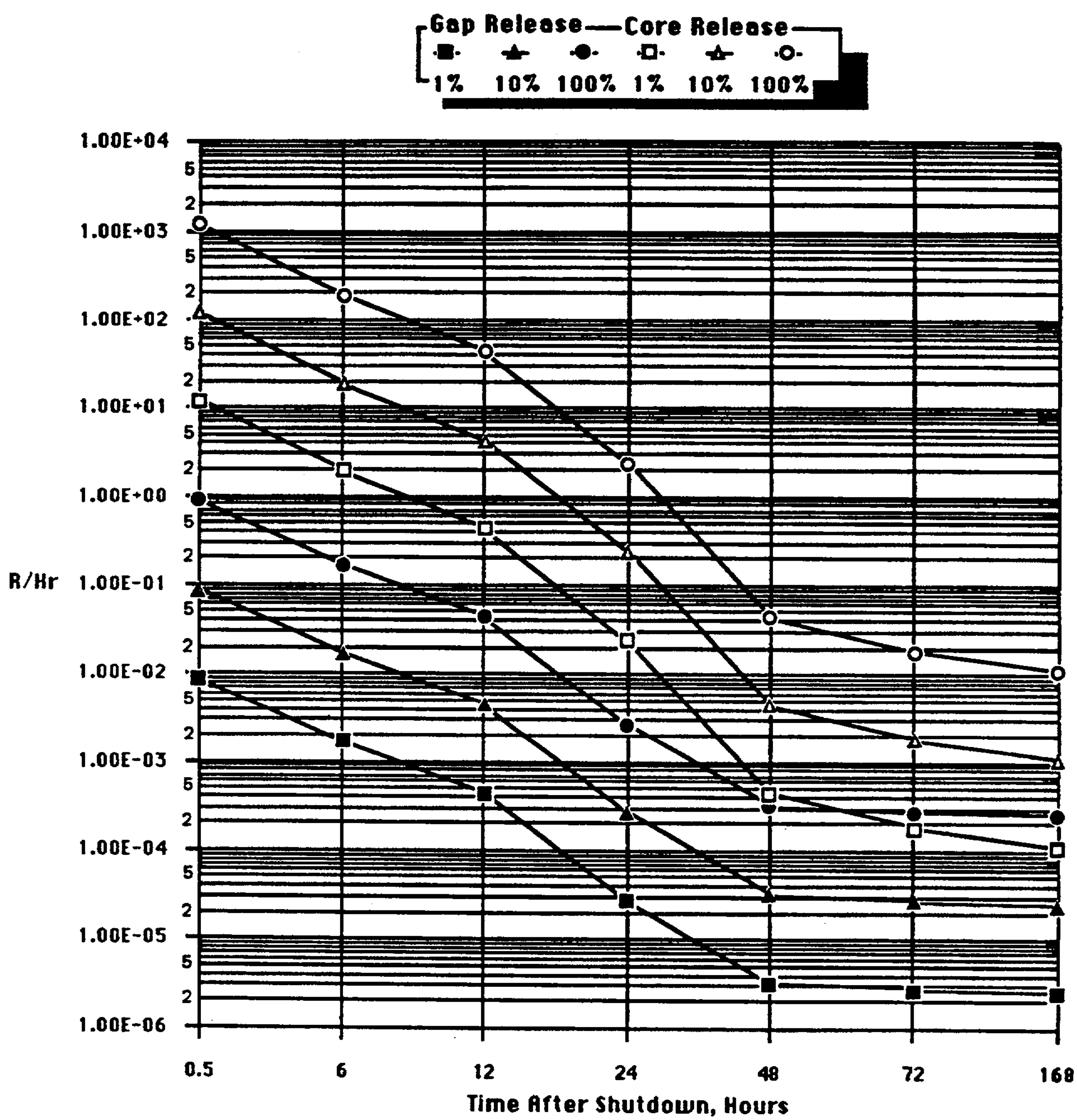


00408204.3Bb 18II

APPENDIX 7.6

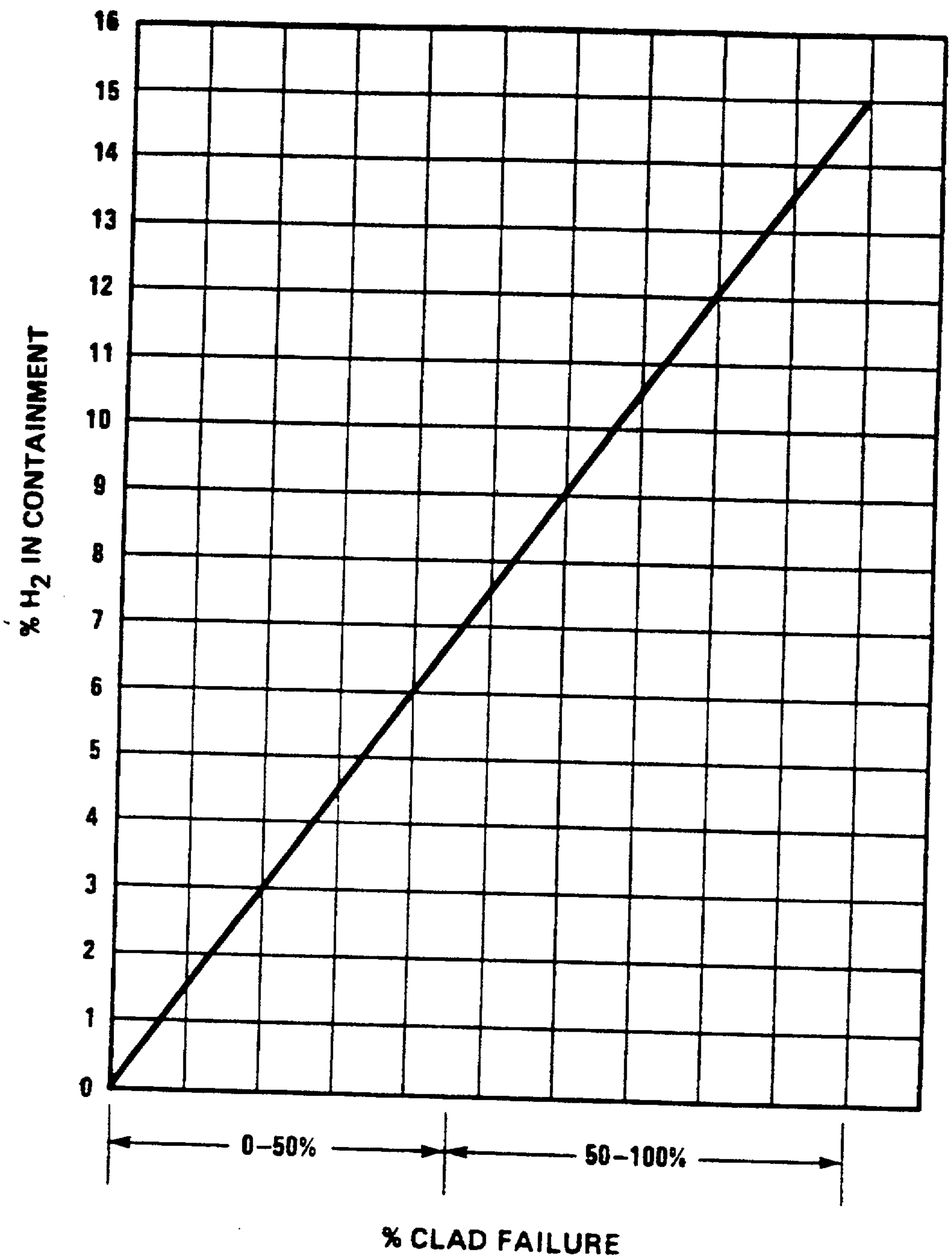
FIGURE 5

POST LOCA EXPOSURE RATE OUTSIDE OF PERSONNEL HATCH FROM NOBLE GASES - GAP AND CORE RELEASE



APPENDIX 7.7

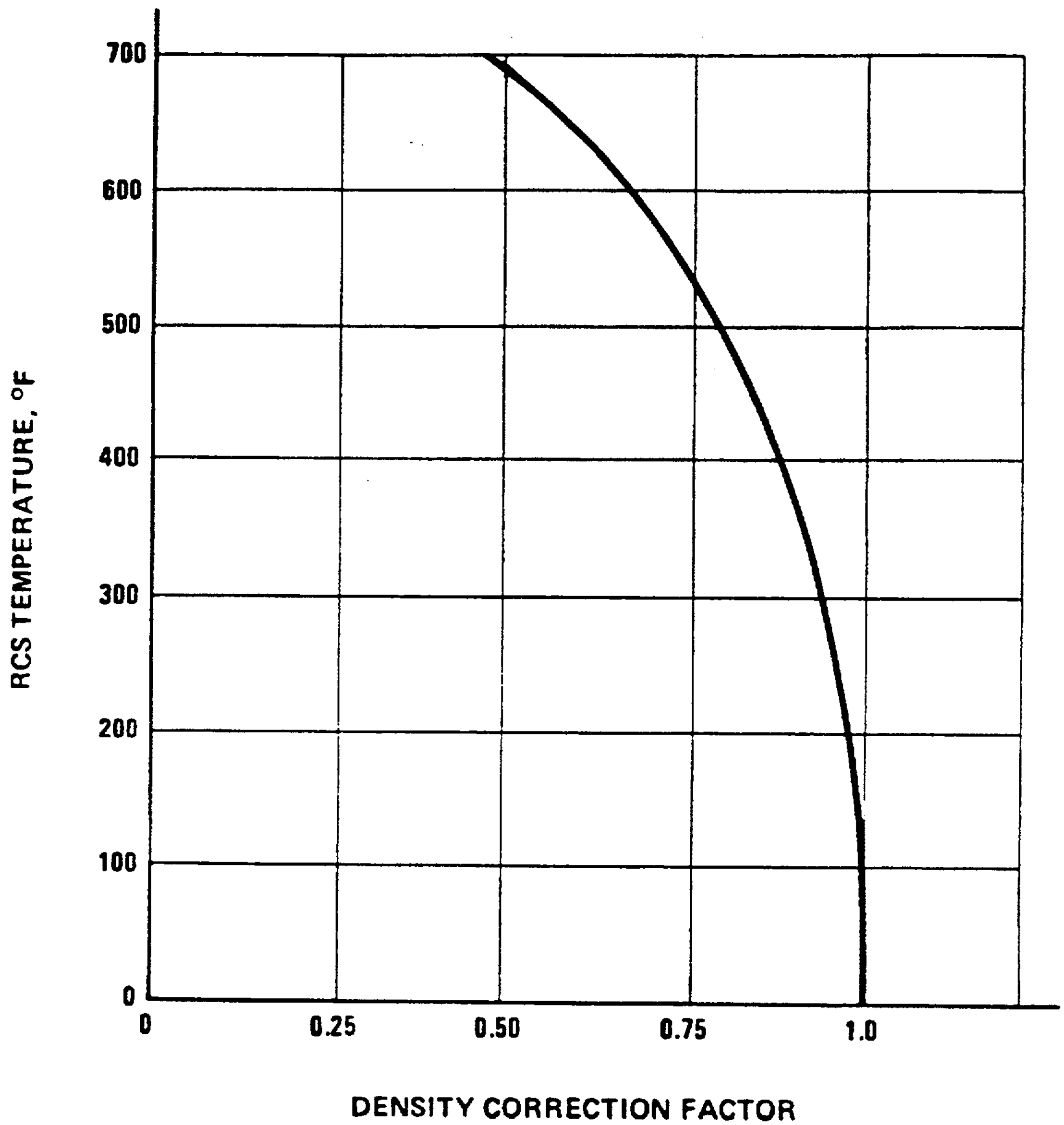
FIGURE 6
PERCENT H2 IN CONTAINMENT VERSUS PERCENT CLAD FAILURE



APPENDIX 7.8

FIGURE 7

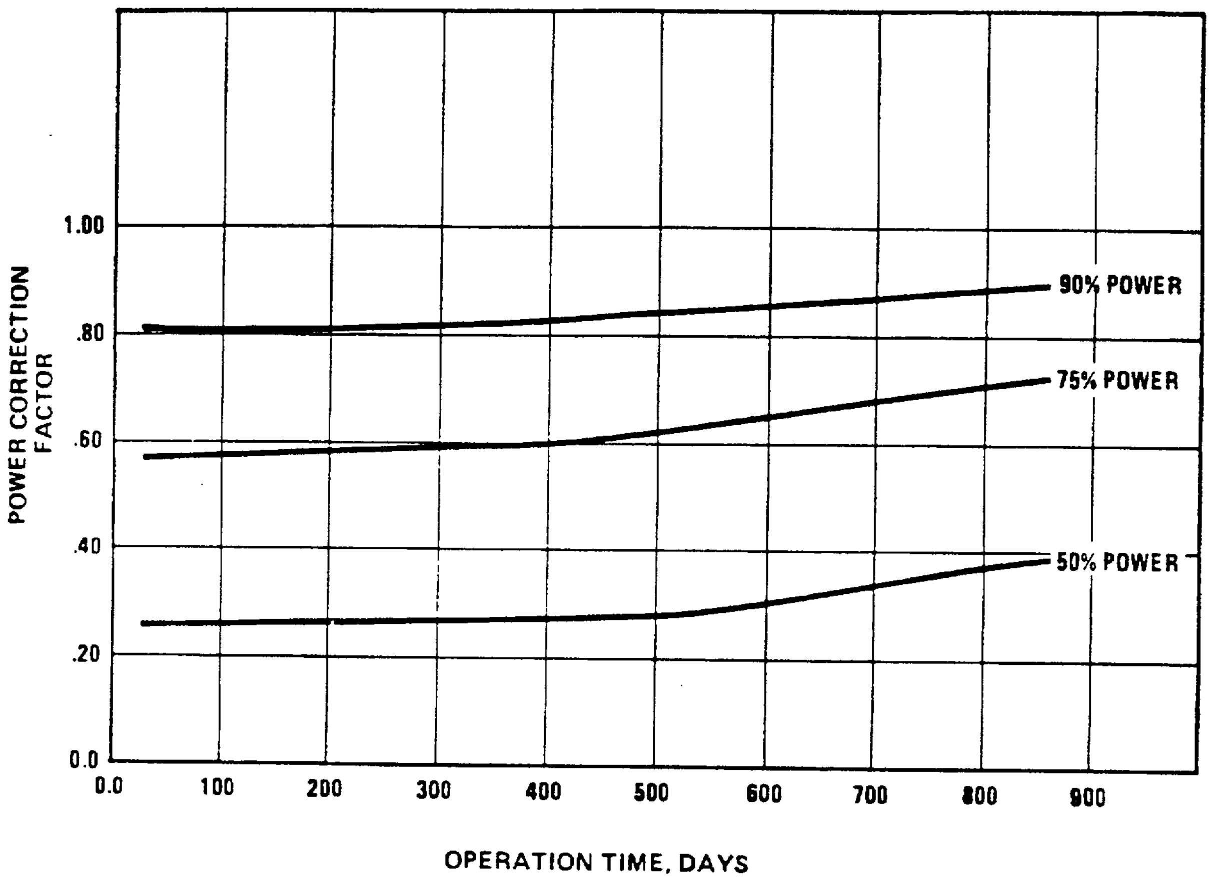
DENSITY CORRECTION FACTOR



APPENDIX 7.9

FIGURE 8

POWER CORRECTION FACTOR FOR CS-134



DIABLO CANYON POWER PLANT
EP RB-14
ATTACHMENT 8.1

1 AND 2

TITLE: Preliminary Assessment

PART 4.1 GENERAL INFORMATION

4.1.1 Date __________ Time ____________ Unit ________ Calculation # ____________

Completed By ____

4.1.2 Time After Reactor Shutdown ____________ Hr Postulated Accident ____________
Table 1

PART 4.2 INADEQUATE CORE COOLING

4.2.1	**INDICATION OF CONDITIONS**	**Yes**	**No**
a.	Are five or more Core Exit Thermocouples (CETC) temperatures greater than 1,200°F?	[ ]	____
b.	Can Safety Injection (SI) and/or charging flow to the Reactor Coolant System (RCS) be verified?	____	[ ]
c.	Can Auxiliary Feed Water (AFW) flow to the steam generators and Component Cooling Water (CCW) and Auxiliary Salt Water (ASW) flow be verified?	____	[ ]
d.	Are RCS pressure and CETC temperature (T hottest) within the "Acceptable Area" of subcooling as determined by using Figure 1?	____	[ ]
e.	Are containment area radiation monitor (RE-30 or RE-31) reading greater than 1 R/hr?	[ ]	____
f.	Is containment pressure greater than 1.3 psig?	[ ]	____
g.	Is containment temperature greater than 120°F?	[ ]	____
h.	Is containment hydrogen level as indicated by monitors CEL-82 or CEL-83 up scale?	[ ]	____

4.2.2 **EVALUATION OF CONDITIONS**

If any of the boxes (as opposed to line spaces) in Step 4.2.1 were checked, proceed to Section 4.3.

TITLE: Preliminary Assessment

PART 4.3 CONTAINMENT RADIATION LEVELS

4.3.1 **CONTAINMENT AREA RADIATION MONITORS OPERABLE**

A RE-30 READING	B RE-31 READING	C READINGS PRODUCT	D AVERAGE READING*
R/hr	R/hr	(R/hr)^2	R/hr
		A & B	✔ ____ C

* Use the operable monitor reading as the average reading if only one monitor is operable.

OR

4.3.2 **CONTAINMENT AREA RADIATION MONITOR INOPERABLE**

E Equipment Hatch Reading*		E Personnel Hatch Reading*
R/hr	OR	R/hr

* Use portable ionization chamber.

4.3.3 **PERCENT CLAD AND/OR CORE FAILURE ESTIMATE**

Containment Area Radiation Monitors Operable

F 100% Gap Release*	G Percent Clad Failure	H 100% Core Release*	I Percent Core Failure
R/hr	R/hr	(R/hr)^2	R/hr
Figure 2	**D/F** x 100	Figure 3	**D/H** x 100

OR **K** if Monitors inoperable

M if monitors inoperable

OR

Containment Area Radiation Monitors Inoperable

J 100% Gap Release*	K Percent Clad Failure	L 100% Core Release*	M Percent Core Failure
R/hr	%	(R/hr)^2	%
Figure 4 or 5	**E/J** x 100	Figure 4 or 5	**E/L** x 100

N

Equivalent RE-30/31 Reading

____

F X G OR **H x I**

*Step 4.1.2 Time After Reactor Shutdown

PART 4.4 CONTAINMENT HYDROGEN LEVEL

If Core Failure was indicated in Step 4.3, proceed to Step 5.1.

4.4.1 **PERCENT CLAD FAILURE ESTIMATE**

A CEL-82 READING	B CEL-83 READING	C AVERAGE READING	D PERCENT CLAD FAILURE
%	%	%	%
Figure 4 or 5	E/J x 100	(A + B)/2	Figure 6

*Use the operable monitor reading as the Average Reading if only monitor is operable

TITLE: Long Term Assessment

Pacific Gas and Electric Company
Long-Term Assessment
Diablo Canyon Power Plant Unit Nos. 1 and 2

69-10423 (Rev. 4/90)
Diablo Canyon Power Plant

PART 5.1 GENERAL INFORMATION

5.1.1 Date ____/____/____ Time ____________ Unit ____________ Calculation # ________

5.1.2 Time After Reactor Shutdown ____________ Hr Postulated Accident ____________ *Table 1*

PART 5.2 EMERGENCY CORE COOLING SYSTEM (ECCS) VOLUME INJECTED

5.2.1 REACTOR COOLANT SYSTEM (RCS) SAMPLE*

Check One: Hot Leg 1 ______ Hot Leg 4 ______ Reactor Cavity Sump ______

**Use table 1 and Step 4.1.2 postulated accident to select sampling location.*

5.2.2 DENSITY CORRECTION FACTOR

RCS Temperature	Density Correction Factor
____________ °F	____________ *Figure 7*

5.2.3 DILUTION VOLUME

Prior RWST Volume		Current RWST Volume		Conversion Factor		A Injected RWST Volume
[________ gal	-	________ gal]	x	3,785 cc/gal	=	________ cc

Accumulator Quantity		B Injected Accumulator Volume
________ Accumlators Injected x 4.28E7 cc	=	________ cc

C
RCS Volume
3.56 E8 cc

Dilution Volume = ____________ cc
A + B + C

TITLE: Long Term Assessment

PART 5.3 LIQUID INVENTORY

Isotope	*A* Sample Activity Concentration*	*B* Density Correction Factor	*C* DilutionVolume	*D* Conversion Factor	*E* Liquid Inventory
Kr-87	________ µCi/cc				________ Ci *A x B x C x D*
Xe-133	________ µCi/cc				________ Ci *A x B x C x D*
I-131	________ µCi/cc				________ Ci *A x B x C x D*
I-133	________ µCi/cc	________ *5.2.2*	________ cc *5.2.3*	1E-6 Ci/µCi	________ Ci *A x B x C x D*
Te-132	________ µCi/cc				________ Ci *A x B x C x D*
Ba-140	________ µCi/cc				________ Ci *A x B x C x D*
La-140	________ µCi/cc				________ Ci *A x B x C x D*
Cs-134	________ µCi/cc				________ Ci *A x B x C x D*

**Includes Off Gases.*

PART 5.4 PRESSURE AND TEMPERATURE CORRECTION FACTOR

5.4.1 **CONTAINMENT ATMOSPHERE SAMPLE***

Check One: Containment Air ________ Other ________

**Use Table 1 and Step 4.1.2 postulated accident to select sampling location.*

5.4.2 **PRESSURE AND TEMPERATURE CORRECTION FACTOR**

F Containment Atmosphere Pressure	*G* Sample Pressure	*H* Sample Temperature	*I* Containment Atmosphere Temperature
________ psia *psig + 14.7*	________ psia *psig + 14.7*	________ °R *°F + 460*	________ °R *°F + 460*

J Pressure Ratio	*K* Temperature Ratio	*L* Pressure Temperature Correction Factor
________ *F / G*	________ *H / I*	________ *J x K*

PART 5.5 GASEOUS INVENTORY

Isotope	*M* Sample Activity Concentration	*N* Pressure Temperature Correction Factor	*O* Containment Volume	*P* Conversion Factor	*Q* Gaseous Inventory
Kr-87	________ µCi/cc				________ Ci *M x N x O x P*
Xe-133	________ µCi/cc				________ Ci *M x N x O x P*
I-131	________ µCi/cc	________ *5.4.2 L*	7.36E10 cc	1E-6 Ci/µCi	________ Ci *M x N x O x P*
I-133	________ µCi/cc				________ Ci *M x N x O x P*

TITLE: Long Term Assessment

PART 5.6 INVENTORY CORRECTION—CONSTANT POWER HISTORY

5.6.1 POWER LEVEL*

A Power Level$_1$	*B* Power Level$_2$	Operation Time	Average Power Level
________ %	________ %	________ days	________ %
Prior 4 days	*Prior 30 days*		*Operation Time*

**Power level within ± 10%, otherwise proceed to Step 5.7.*

5.6.2 POWER CORRECTION FACTOR

C Power Correction Factor$_1$	*D* Power Correction Factor$_2$	*E* Cs-134 Power Correction Factor*
________	________	________
A / 100	*B / 100*	*Figure 8*

**Use Step 5.6.1 average power level and operation time.*

5.6.3 CORRECTED INVENTORY—GAP

Isotope	*F* Gap Equilibrium Inventory	*G* Gap Corrected Inventory
Kr-87	3.9E4 Ci	________ Ci (*C x F*)
I-133	5.1E5 Ci	________ Ci (*C x F*)
Xe-133	1.3E6 Ci	________ Ci (*D x F*)
I-131	8.0E5 Ci	________ Ci (*D x F*)

5.6.4 CORRECTED INVENTORY—CORE

Isotope	*H* Core Equilibrium Inventory	*I* Core Corrected Inventory
Kr-87	5.9E7 Ci	________ Ci (*C x H*)
I-133	1.9E8 Ci	________ Ci (*C x H*)
Xe-133	1.9E8 Ci	________ Ci (*D x H*)
Te-132	1.4E8 Ci	________ Ci (*D x H*)
Ba-140	1.8E8 Ci	________ Ci (*D x H*)
La-140	1.8E8 Ci	________ Ci (*D x H*)
I-131	9.7E7 Ci	________ Ci (*D x H*)
Cs-134	3.1E6 Ci	________ Ci (*E x H*)

TITLE: Long Term Assessment

Pag? 4 of 8

PART 5.7 INVENTORY CORRECTION—VARIABLE POWER HISTORY

5.7.1 **POWER LEVEL**

Operation Time	Average Power Level
_____________ days	_____________________ %
	Operation Time

5.7.2 **POWER CORRECTION FACTOR**

Cs-134
Power Correction Factor*

____
Figure 8

Use the following equation to determine the power correction factor for other isotopes.

$$PF_i = \frac{\Sigma jPj\,(1-e^{-\lambda_i t_{1j}})e^{-\lambda_i t_{2j}}}{RP}$$

Where:

PF_i = 30 day Power Correction Factor for nuclide i,
Pj = average power level (Mwt) for period t_{1j},
λ_i = decay constant (days -1) for nuclide i,
t_{1j} = time (days) where power does not vary more than ±10% from Pj,
t_{2j} = time (days) from end of period t_{1j} to time of reactor shutdown,
RP = rated power level of the core (Mwt).

**Use Step 5.7.1 operation time and average power level.*

Page 5 of 8

Isotope	A λ_i	B t_{1j}	C t_{2j}	D $\lambda_i t_{1j}$	E $\lambda_i t_{2j}$
		______ days	______ days (30 - B)	______ (A x B)	______ (A x C)
		______ days	______ days (30 - B)	______ (A x B)	______ (A x C)
______	______ $days^{-1}$	______ days	______ days (30 - B)	______ (A x B)	______ (A x C)
		______ days	______ days (30 - B)	______ (A x B)	______ (A x C)
		______ days	______ days (30 - B)	______ (A x B)	______ (A x C)
		______ days	______ days (30 - B)	______ (A x B)	______ (A x C)

F $e^{-\lambda_i t_{1j}}$	G $1 - e^{-\lambda_i t_{1j}}$	H $e^{-\lambda_i t_{2j}}$	I P_j*	J Correction Factor
______ (EXP [-D])	______ (1 - F)	______ (EXP [-E])	______ Mwt	______ (G x H x I)
______ (EXP [-D])	______ (1 - F)	______ (EXP [-E])	______ Mwt	______ (G x H x I)
______ (EXP [-D])	______ (1 - F)	______ (EXP [-E])	______ Mwt	______ (G x H x I)
______ (EXP [-D])	______ (1 - F)	______ (EXP [-E])	______ Mwt	______ (G x H x I)
______ (EXP [-D])	______ (1 - F)	______ (EXP [-E])	______ Mwt	______ (G x H x I)
______ (EXP [-D])	______ (1 - F)	______ (EXP [-E])	______ Mwt	______ (G x H x I)

Correction Factor Σ = ______

*Power level for period t_{1j}.

Correction Factor		Rated Power Level*		Power Correction Factor
______	÷	______ Mwt	=	______

*Use 3338 Mwt for Unit 1 or 3411 for Unit 2.

Page 6 of 8

5.7.3 CORRECTED INVENTORY—GAP

Isotope	Decay Constant	*A* Power Correction Factor	*B* Gap Equilibrium Inventory	*C* Gap Corrected Inventory
Kr-87	1.31E 1 days^{-1}	______ 5.7.2	3.9E4 Ci	______ Ci *A x B*
Xe-133	1.3E-1 days^{-1}	______ 5.7.2	1.3E6 Ci	______ Ci *A x B*
I-131	8.62E-2 days^{-1}	______ 5.7.2	8.0E5 Ci	______ Ci *A x B*
I-133	8.00E-1 days^{-1}	______ 5.7.2	5.1E5 Ci	______ Ci *A x B*

5.7.4 CORRECTED INVENTORY—CORE

Isotope	Decay Constant	*D* Power Correction Factor	*E* Core Equilibrium Inventory	*F* Core Corrected Inventory
Kr-87	1.31E 1 days^{-1}	______ 5.7.2	5.9E7 Ci	______ Ci *D x E*
Xe-133	1.3E-1 days^{-1}	______ 5.7.2	1.9E8 Ci	______ Ci *D x E*
Te-132	2.13E-1 days^{-1}	______ 5.7.2	1.4E8 Ci	______ Ci *D x E*
Ba-140	5.42E-2 days^{-1}	______ 5.7.2	1.8E8 Ci	______ Ci *D x E*
La-140	4.1E-1 days^{-1}	______ 5.7.2	1.8E8 Ci	______ Ci *D x E*
I-131	8.62E-2 days^{-1}	______ 5.7.2	9.7E7 Ci	______ Ci *D x E*
I-133	8.00E-1 days^{-1}	______ 5.7.2	1.9E8 Ci	______ Ci *D x E*
Cs-134	9.24E-4 days^{-1}	______ 5.7.2	3.1E6 Ci	______ Ci *D x E*

TITLE: Long Term Assessment

Page 7 of 8

PART 5.8 RELEASE PERCENT, IODINE RATIO, AND NOBLE GAS RATIO

5.8.1 **GAP RELEASE PERCENT**

Isotope	*A* Liquid Inventory	*B* Gaseous Inventory	*C* Released Inventory	*D* Corrected Inventory	*E* Release Percent
Kr-87	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.3 or 5.7.3)	____ (C / D x 100)
Xe-133	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.3 or 5.7.3)	____ (C / D x 100)
I-131	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.3 or 5.7.3)	____ (C / D x 100)
I-133	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.3 or 5.7.3)	____ (C / D x 100)

5.8.2 **CORE RELEASE PERCENT**

Isotope	*A* Liquid Inventory	*B* Gaseous Inventory	*C* Released Inventory	*D* Corrected Inventory	*E* Release Percent
Kr-87	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.4 or 5.7.4)	____ (C / D x 100)
Xe-133	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.4 or 5.7.4)	____ (C / D x 100)
I-131	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.4 or 5.7.4)	____ (C / D x 100)
I-133	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.4 or 5.7.4)	____ (C / D x 100)
Te-132	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.4 or 5.7.4)	____ (C / D x 100)
Ba-140	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.4 or 5.7.4)	____ (C / D x 100)
La-140	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.4 or 5.7.4)	____ (C / D x 100)
Cs-134	____ Ci (5.3)	____ Ci (5.5)	____ Ci (A + B)	____ Ci (5.6.4 or 5.7.4)	____ (C / D x 100)

5.8.3 **IODINE RATIO**

F I-133 Released Inventory	*G* I-131 Released Inventory	Iodine Ratio
____ Ci (5.8.1 C)	____ Ci (5.8.1 C)	____ (F ÷ G)

5.8.4 **NOBLE GAS RATIO**

H Kr-87 Released Inventory	*I* Xe-133 Released Inventory	Noble Gas Ratio
____ Ci (5.8.1 C)	____ Ci (5.8.1 C)	____ (H ÷ I)

Page 8 of 8

PART 5.9 ASSESSMENT WORKSHEET

5.9.1 **ASSESSMENT WORKSHEET**

	Clad Failure*			**Core Failure***					
				Overheat			**Melt**		
Isotope	**0.12 - 10%**	**10 - 50%**	**>50%**	**0.12 - 10%**	**10 - 50%**	**>50%**	**0.12 - 10%**	**10 - 50%**	**>50%**
Kr-87									
Xe-133									
I-131									
I-133									
Iodine Ratio**	$\frac{\text{I-133}}{\text{I-131}} < .64$			$.64 \le \frac{\text{I-133}}{\text{I-131}} < 1.94$			$\frac{\text{I-133}}{\text{I-131}} \ge 1.94$		
Noble Gas Ratio**	$\frac{\text{Kr-87}}{\text{Xe-133}} < .03$			$.03 \le \frac{\text{Kr-87}}{\text{Xe-133}} < .31$			$\frac{\text{Kr-87}}{\text{Xe-133}} \ge .31$		
			Cs-134						
			Te-132						
						Ba-140			
						La-140			

Percent Clad Failure: ________ %

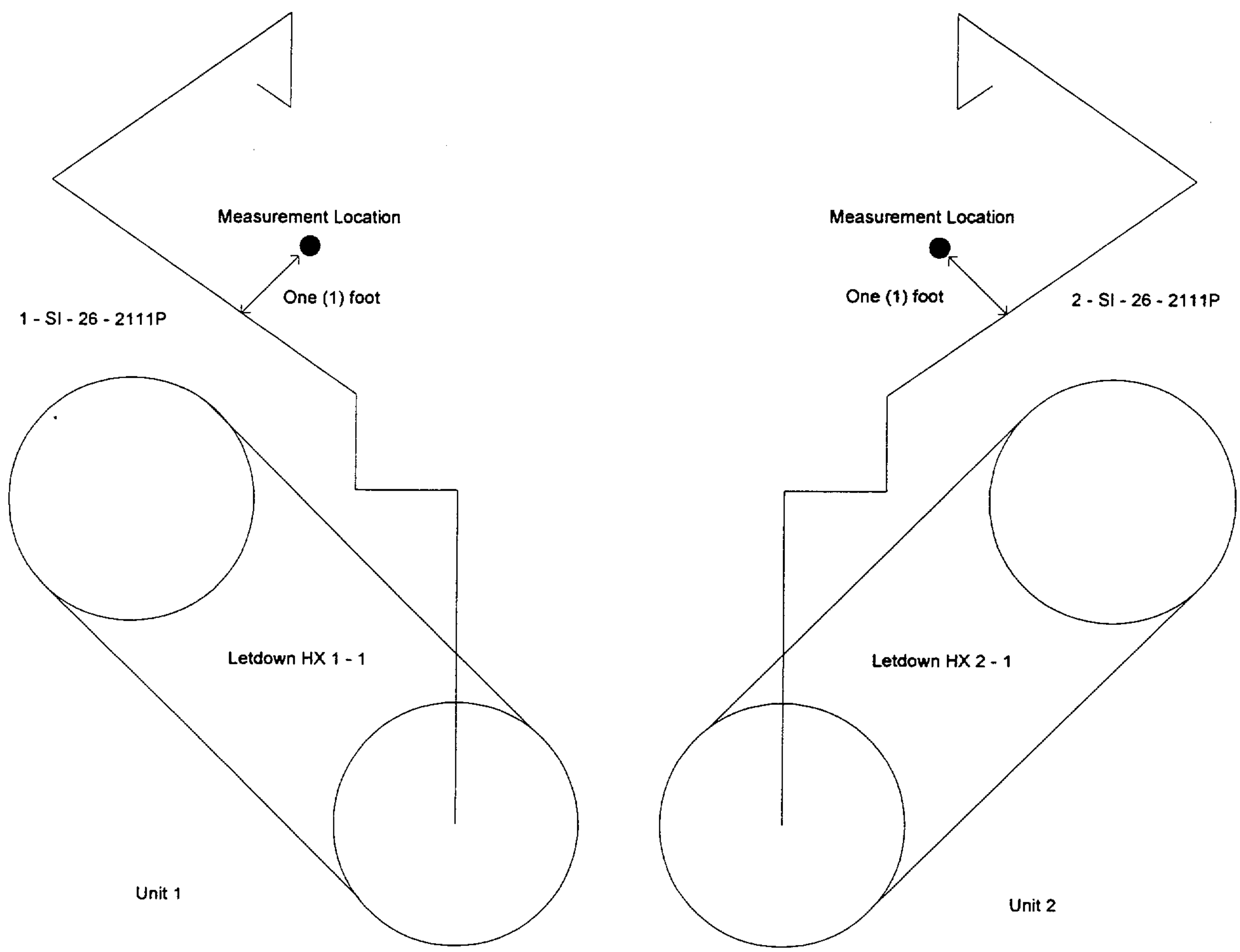
Percent Core Failure: ________ %

*Steps 5.8.1 and 5.8.2 Release Percent
**Steps 5.8.3 and 5.8.4 Iodine and Noble Gas Ratios

DIABLO CANYON POWER PLANT
EP RB-14
ATTACHMENT 8.3

1 AND 2

TITLE: Initial Detection of Potential Core Damage



Measure and record the radiation level one (1) foot from the center of line S1-26-2IIIP. A radiation level equal to or greater than 15 R/hr indicates core damage at the Alert emergency action level.