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

DEPARTMENT OF NUCLEAR PLANT OPERATIONS

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

TITLE. EMERGENCY PROCEDURE
CORE DAMAGE ASSESSMENT PROCEDURE

NUMBER EP RB-14
REVISION 0
DATE 4/7/84
PAGE 1 OF 11

IMPORTANT TO SAFETY

APPROVED: ______________________________
PLANT MANAGER DATE

## SCOPE

This procedure describes the evaluation of the extent of core damage following an accident that can lead to inadequate core cooling. This procedure and revision thereto requires PSRC review.

## DISCUSSION

Fuel damage resulting in the release of radioactive material can occur following a loss of coolant accident (LOCA) or following the loss of available heat sinks. These events, if uncorrected, can lead to localized or widespread overheating of reactor fuel and eventually to fuel rod cladding failure, and/or fuel melt. The description of plant parameters indicative of conditions that can lead to fuel failure or melt are provided in the Emergency Equipment Operation Procedures (OP Series).

This procedure supplements other emergency procedures by providing a methodology to determine the extent of core damage that may have resulted from an accident. This procedure does not replace procedures that are used to provide instructions regarding accident identification and/or mitigation or dose assessment although similar data and assessment is utilized in this procedure to determine the type and extent of fuel damage.

The objective of this procedure is the classification of fuel damage into one of four broad categories: (1) no fuel damage, (2) fuel cladding damage, (3) fuel overheat, and (4) fuel melt. Within the latter three categories, the procedure permits a rough estimate of damage as a proportion of core radionuclide inventories that have been released to the reactor coolant and/or containment atmosphere.

It provides a preliminary and a long term methodology for assessing core damage. The preliminary assessment utilizes rough evaluations of plant parameters such as reactor vessel level and reactor coolant temperatures to confirm that conditions exist which can lead to core damage, and quantifies the damage through the use of containment hydrogen levels and containment radiation levels. The long-term methodology requires that reactor coolant and containment air samples



8405170118 840511
PDR ADOCK 05000275
P PDR

be obtained and analyzed for radiochemical and chemical parameters. These results are then used, along with other plant parameters, to determine the extent of core damage. The preliminary assessment can yield quick initial results (within approximately 10 minutes). The long-term assessment could require up to three (3) hours to obtain sample analyses and further time for computation and evaluation of the results, but yields additional information necessary to distinguish between cladding failures, fuel overheat, and fuel melt.

PROCEDURE

1. Preliminary Assessment

   a. Corroborating Evidence for Core Damage

      If evaluation personnel in the TSC have indicated that conditions exist that could lead to inadequate core cooling (per EP OP-0 Appendix B), or ineffective ESF functioning under LOCA conditions (per EP OP-1, Appendix H), skip this section and proceed to Section b. Otherwise, determine the potential for fuel damage as described below:

Check the Appropriate Answer

| | | YES | NO |
|---|---|---|---|
| 1) | Are five or more core exit thermocouples temperatures greater than 1,200°F? | [ ] | ___ |
| 2) | Can SI and/or charging flow to the RCS be verified? | ___ | [ ] |
| 3) | Can AFW flow to the steam generators and CCW and ASW flow be verified? | ___ | [ ] |
| 4) | Are RCS pressure and temperature within the "Acceptable Area" of subcooling as determined using Figure 1? | ___ | [ ] |
| 5) | Are containment rad. monitors (RE-30 and 31) reading greater than 1R/hr.? | [ ] | ___ |
| 6) | Is containment pressure greater than 1.3 psig? | [ ] | ___ |

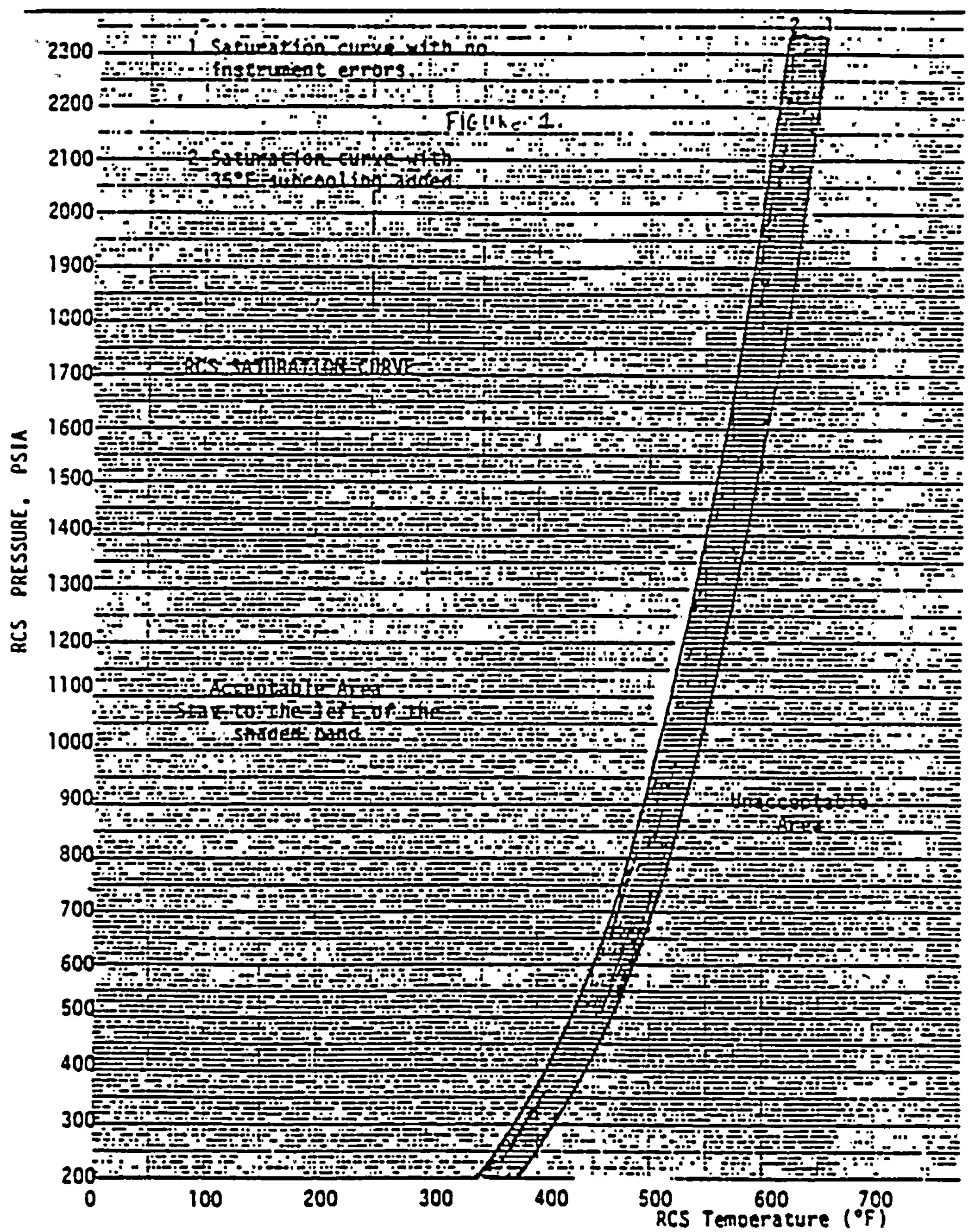
| | YES | NO |
|---|---|---|
| 7) Is containment temperature greater than 120°F? | [ ] | ___ |
| 8) Is containment hydrogen level as indicated by monitors CEL-82 and CEL-83 up scale? | [ ] | ___ |

If any of the boxes (as opposed to line spaces) for the previous questions were checked, then those conditions are potentially indicative of inadequate core cooling or a LOCA*, therefore, continue with this procedure. If none of the boxes were checked, continue monitoring the situation in accordance with applicable procedures.

*NOTE: In general, the more boxes that are checked, the greater the potential for inadequate core cooling. However, evaluation is necessary to determine the significance of this information. For conservatism, continue with this procedure if a full evaluation cannot be performed.

| DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2 | NUMBER EP RB-14<br>REVISION 0<br>DATE 4/7/84<br>PAGE 4 OF 11 |
|---|---|
| TITLE: CORE DAMAGE ASSESSMENT PROCEDURE | |

FIGURE 1



b. Preliminary Assessment of LOCA resulting in Core Damage

If loss of reactor coolant to the containment is not occurring, skip this section and proceed to Section 2.

1) Record time since reactor trip. _________ hrs A

2) Record reading of containment area monitor (RE-30). _________ R/hr B

3) Record reading of containment area monitor (RE-31). _________ R/hr C

4) Determine average area monitor reading [(B + C)/2]. _________ R/hr D

5) If the value of Item D is < 1 R/hr, then no core damage is indicated, otherwise compare D to calculated curves of area monitor response provided in Figures 2 and 3. Estimate the level of clad or core damage by interpolating between the curves provided for the time interval A. Record the results as indicated.

NOTE: Interpolation can be accomplished by taking the reading D and dividing by the expected monitor reading for 100 percent core melt or GAP release at time A, multiplied by 100.

Check One
___ No damage
___ GAP ___%
___ Melt

| DIABLO CANYON POWER PLANT UNIT NO(S) 1 AND 2 | NUMBER EP RB-14<br>REVISION 0<br>DATE 4/7/84<br>PAGE 6 OF 11 |
|---|---|
| TITLE. CORE DAMAGE ASSESSMENT PROCEDURE | |

FIGURE 2
EXPOSURE RATE VS POST ACCIDENT DECAY TIME
RE 30 AND 31 WITH GAP RELEASES

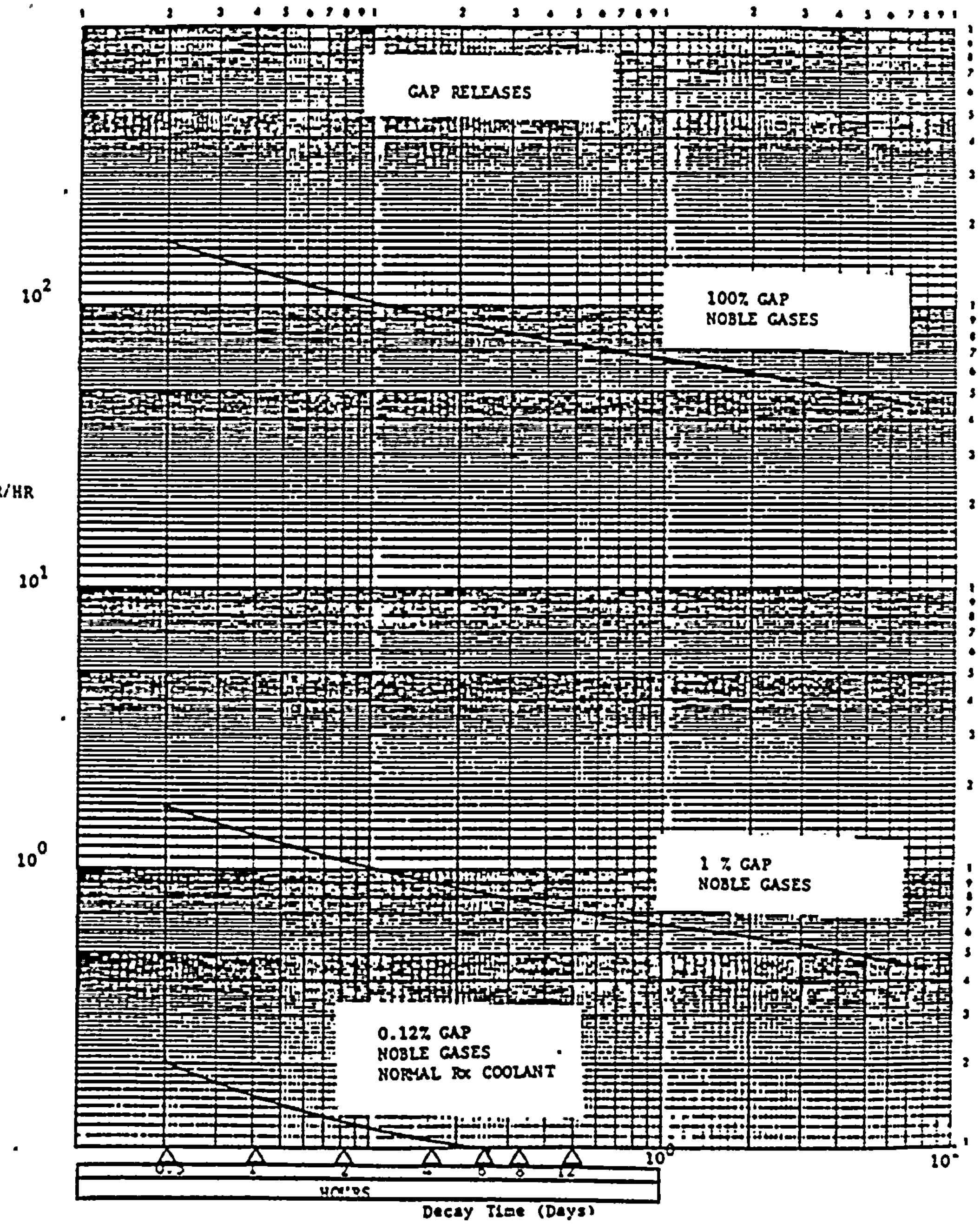
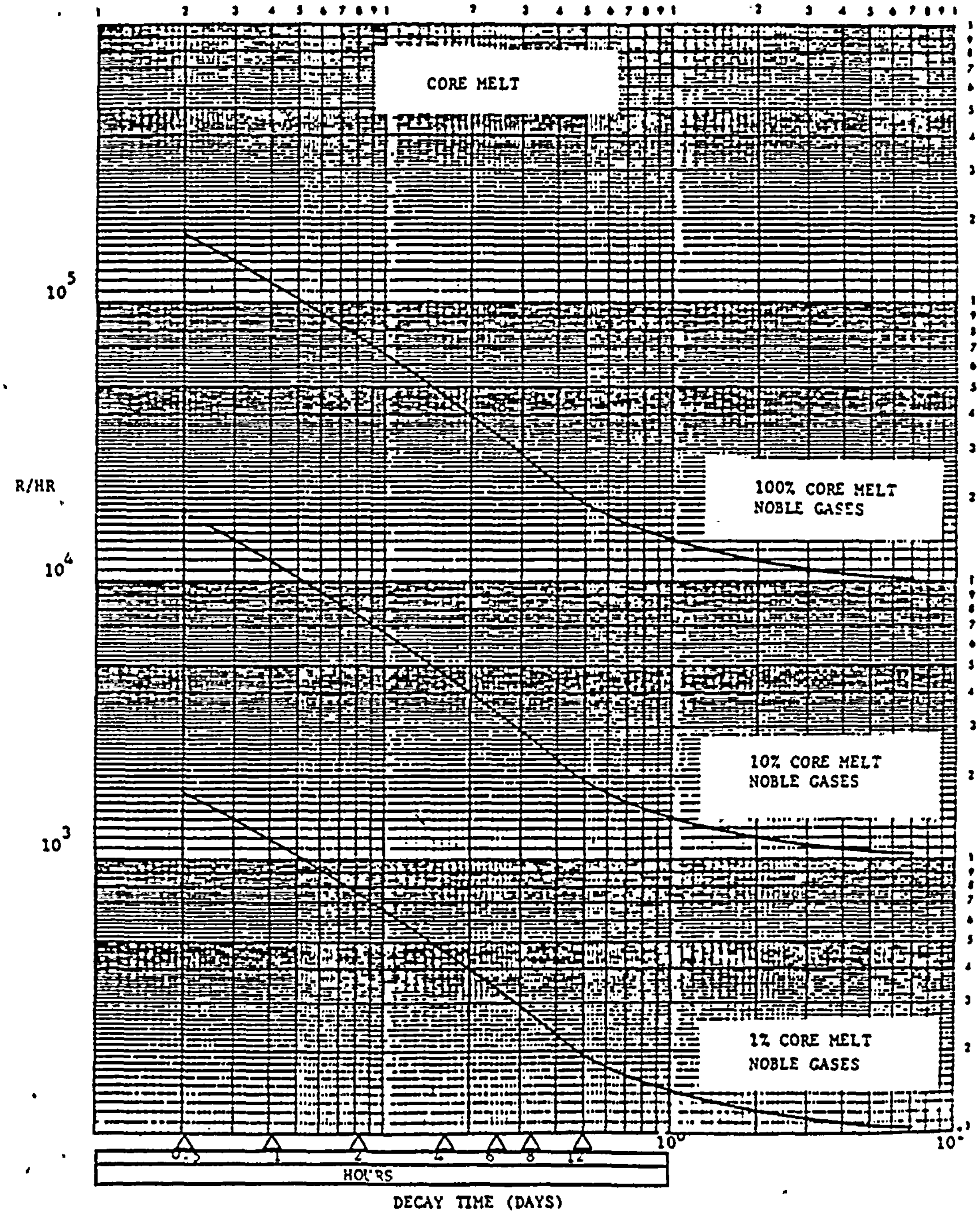


FIGURE 3
EXPOSURE RATE VS POST ACCIDENT DECAY TIME
RE 30 AND 31 CORE MELT



6) If no damage, or a GAP release is indicated, then corroborative results can be attempted using containment hydrogen levels. If core melt is indicated, proceed to Section 2 of this procedure.

7) Determine the average containment hydrogen concentration from containment hydrogen monitors CEL-82 and CEL-83. ________ % E

8) Compare Item E determined in Step 7 with the curve of expected $H_2$ concentration versus percent clad failure found in Figure 4. Estimate the percent clad damage and record the result.+

Check One
___No damage
___Clad failure ___%

2. Long-Term Assessment

a. Request Sample

1) Request that the Emergency Radiological Advisor (ERA) assign a sampling team to collect a RCS and containment air sample from the post-accident sampling system (PASS). The following analyses shall be performed:

a) Gamma spectrometry on:
(1) RCS liquid and off-gas
(2) Containment air

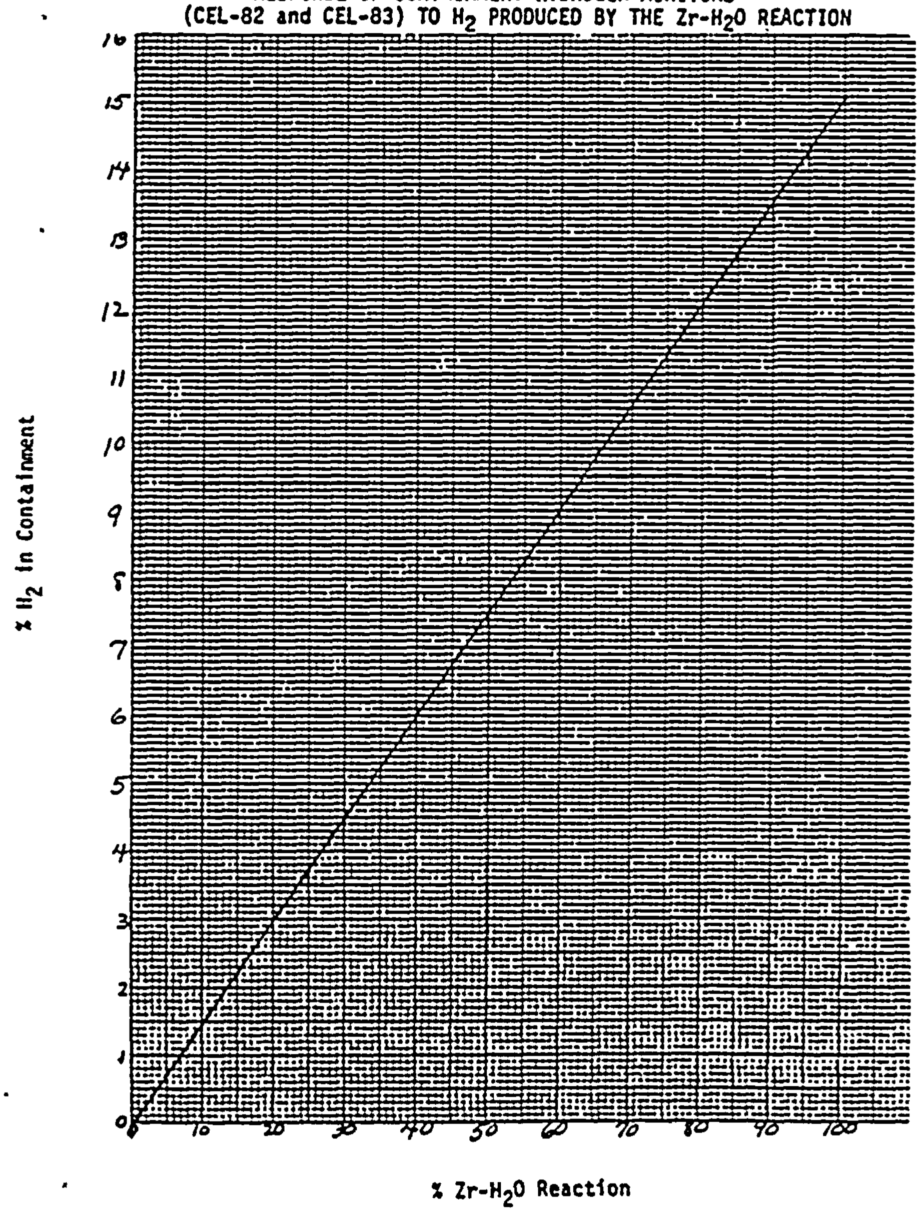
+NOTE: Hydrogen levels in containment are a valid indicator of damage only within the first 24 hours of the accident, assuming that the hydrogen recombiners are not operating. Since complex mechanisms dictate the amount of $H_2$ and radioactive materials released to containment it is not possible to predict which assessment of clad damage is more accurate. If results of damage assessment using the rad. monitors and the $H_2$ monitors differ, try to utilize corroborating data from RVLIS, etc., to select the most representative assessment of damage. If resolution cannot be obtained, use the highest estimated level of clad failure.

b) $H_2$ levels on:
(i) Containment air

Request that the sample analysis be decay corrected back to the time of sampling.

FIGURE 4

RESPONSE OF CONTAINMENT HYDROGEN MONITORS
(CEL-82 and CEL-83) TO $H_2$ PRODUCED BY THE Zr-$H_2$O REACTION



2) Await sample analyses. This may take up to 3 hours. In the interim, continue to assess core damage via the techniques provided in Part 1 of this procedure. When sample analyses are available, proceed to Section 2.2.

b. Analysis of Sample Results

1) Complete Attachment 1, "Water Entrained Inventory Worksheet."

2) Complete Attachment 2, "Airborne Inventory Worksheet."

3) If reactor power for the 30 days prior to shutdown remained relatively constant (within the range of ±20 percent) and the average power level is at least 80 percent, proceed to Attachment 3 and determine the power corrected source inventories. Otherwise, complete Attachment 4 to correct source inventories for a variable power history.

4) Complete Attachment 5 to determine percentage of core or GAP activity released.

5) Refer to Attachment 6 to determine what type of failure (no damage, clad, overtemp, melt) is occurring (if any).

6) Report results determined by this procedure to the individual making the reauest for a core damage assessment.

7) Continue to monitor the situation by utilizing this procedure as necessary.

ATTACHMENTS

1. Water Entrained Inventory Worksheet
2. Airborne Inventory Worksheet
3. Calculation of Power-Corrected Source Inventories for Constant Power Levels
4. Source Inventory Power Correction for Variable Power History
5. Comparison of Expected and Actual Source Inventories
6. Qualitative Assessment

TITLE: WATER ENTRAINED INVENTORY WORKSHEET

ATTACHMENT 1

1. Check the appropriate sample type:

   ____ RCS ____ RHR ____ Reactor Cavity Sump (A)

2. Convert the elapsed time since reactor trip and the collection of the RCS sample to hours:

   ________________ hr (B)

3. If the sample is a RCS sample, record the following information:

   RCS Temperature (Tave) ________________ °F (C)

   RCS Pressure ________________ psia (D)

   RCS Density Correction Factor (Figure 5) ________________ (E)

4. Last RWST volume prior to accident ________________ gal (F)

5. Current RWST volume ________________ gal (G)

6. Volume of RWST injected ((F)-(G)) = ________________ gal (H)

7. Determine which ECCS volumes have been released into the RCS or containment, and determine the total volume in the Containment Sump:

| Total Volume | Actual Used in RCS | |
|---|---|---|
| (1) RWST Volume = (H) x 3,785 $\frac{cc}{Gal}$ = | ____________ | cc (I) |
| (2) Each Accumulator = 4.28 X $10^7$ cc (there are four accumulators) | ____________ | cc (J) |
| (3) RCS = 3.56 X $10^8$ cc | 3.56 x $10^8$ | cc (K) |
| (4) Total Volume ((I) + (J) + (K)) | ____________ | cc (L) |

8. Enter the values of (E) and (L) on the next page and calculate the total water entrained inventory as indicated.

9. Proceed to Step 2.2.1 of this procedure.

TITLE: WATER ENTRAINED INVENTORY WORKSHEET

ATTACHMENT 1 (Continued)

| | A | B[a] | C[b] |
|---|---|---|---|
| Isotope | Measured Liquid† Sample Activity (μCi/cc) | Activity at RCS Conditions (μCi/cc) | Total Water Entrained Inventory (Ci) |
| Kr-87 | | | |
| Xe-133 | | | |
| I-131 | | | |
| I-132 | | | |
| Te-132 | | | |
| Cs-134 | | | |
| Ba-140 | | | |
| La-140 | | | |

[a] B = A x E ____________

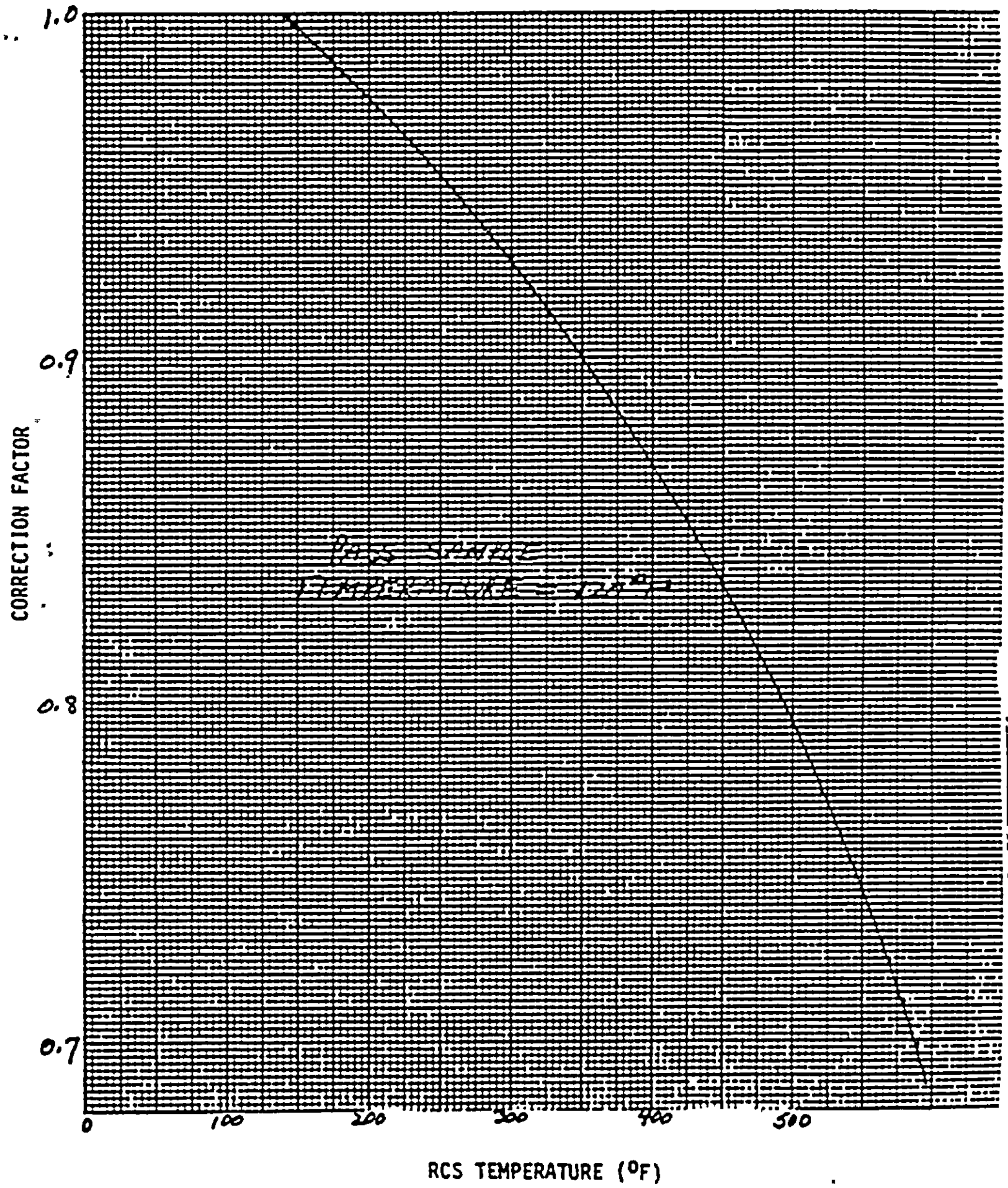
[b] C = B x L ____________ x $10^{-6}$

†Liquid is defined as the total activity from the RCS sample, including any off gases.

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2
FIGURE 5

TITLE: WATER ENTRAINED INVENTORY WORKSHEET

PASS LIQUID SAMPLE DENSITY CORRECTION



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

TITLE: AIRBORNE INVENTORY WORKSHEET

ATTACHMENT 2

1. Record the elapsed decay time since reactor trip and the collections of the containment air sample in hours:

   ________ nr (A)

2. Record the following information:

   Containment Temperature ________ °F (B)
   Containment Pressure ________ psig (C)
   (C) + 14.7 = ________ psia (D)

3. Calculate the Containment atmosphere Pressure and Temperature (P-T) correction factor using the following formula:

$$\text{P-T Correction Factor} = \frac{(D)}{14.7} \cdot \frac{(530)}{((B) + 460)}$$

   Where 530 is room temperature on Rankine Scale.

   P-T Correction Factor ________ (E)

4. Record the containment airborne sample activities in Column A on page 2 of this attachment. (NOTE: The containment activities reported by the sampling teams are at room temperature and pressure.

5. Adjust the reported activities to the conditions of temperature and pressure found in the containment by multiplying the values in Column A by the P-T correction factor, (E) and recording the result in Column B.

6. Calculate the total airborne inventory of the nuclides of interest by multiplying all the values in Column B by 7.36E+04. (This is the Containment volume of 7.36E+10cc and the conversion factor of $10^{-6}$ Ci/μCi.) The results should be recorded in Column C.

TITLE: AIRBORNE INVENTORY WORKSHEET

ATTACHMENT 2 (Continued)

| | A | B[a] | C[b] |
|---|---|---|---|
| Nuclide | Measured Sample Activity (μCi/cc) | Activity at Containment Conditions (μCi/cc) | Total Airborne Inventory (Ci) |
| Kr-87 | | | |
| Xe-133 | | | |
| I-131 | | | |
| I-132 | | | |

[a] B = A x E

[b] C = B x 7.36E+04

7. Proceed to Step 2.b.3)

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

TITLE: CALCULATION OF POWER-CORRECTED SOURCE INVENTORIES FOR CONSTANT POWER LEVELS

## ATTACHMENT 3

1. Estimate and record the average power for the last 30 days. (This form is only to be used when the power level has averaged at least 80 percent and has remained relatively constant [within the range of ±20 percent in the last 30 days]. If the power level has not been relatively constant, use Attachment 4. ____________%

2. Multiply the source inventory values listed in Column 1 by the value on Line 1, and record the result in Column 2.

3. Enter the corrected values.

| NUCLIDE | 1<br>EQUILIBRIUM SOURCE (Ci) | 2<br>CORRECTED SOURCE (Ci) |
|---|---|---|
| **GAP INVENTORY** | | |
| Kr-87 | 3.9E+4 | |
| Xe-133 | 1.3E+6 | |
| I-131 | 8.0E+5 | |
| I-132 | 1.3E+5 | |
| **FUEL PELLET INVENTORY** | | |
| Kr-87 | 5.9E+6 | |
| Xe-133 | 1.9E+8 | |
| Te-132 | 1.4E+8 | |
| Cs-134 | 3.1E+6 | |
| Ba-140 | 1.6E+8 | |
| La-140 | 1.8E+9 | |
| I-131 | 9.7E+7 | |
| I-132 | 1.4E+8 | |

TITLE: SOURCE INVENTORY POWER CORRECTION FOR VARIABLE POWER HISTORY

ATTACHMENT 4

When the power level has not been relatively constant at 80 percent for the last 30 days, nuclides of interest have not had enough time to build up to equilibrium levels.

In this case, the effects of each significant power change must be taken into account. The formula to be used is:

$$PF_i = \Sigma_j F_j \, (1-e^{-\lambda_i t_{1j}}) e^{-\lambda_i t_{2j}}$$

Where:

- $PF_i$ = 30-day power factor for Nuclide i
- $F_j$ = fractional power level for time Period j
- $\lambda_i$ = nuclear decay constant for Nuclide i
- $t_{1j}$ = length of time Period j
- $t_{2j}$ = time from end of time Period j to end of 30-day period.

The calculation of power factor must be repeated for every nuclide.

The power factor determined in this way should be recorded in Column B of this from. This power factor should then be multiplied by the value in Column C with the result recorded in Column D.

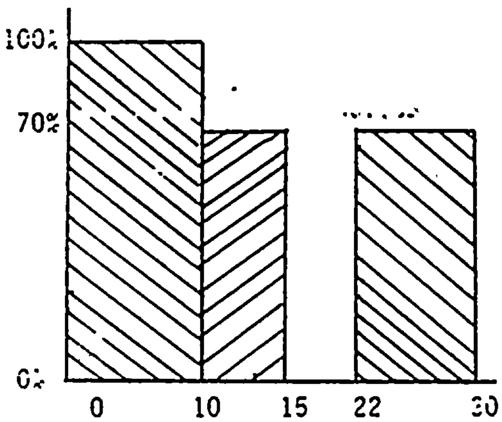
Sample power factor calculation:

The Plant has operated for the 30 days prior to the accident with the following power history:

100-percent power for 10 days
70-percent power for 5 days
0-percent power for 7 days
70-percent power for 8 days

TITLE: SOURCE INVENTORY POWER CORRECTION FOR VARIABLE POWER HISTORY

ATTACHMENT 4 (Continued)




30 Day Power History Prior to Accident

For time Period 1,

$F_1 = 1.0$, $t_{11} = 10$ days, $t_{21} = 20$ days

For time Period 2,

$F_2 = 0.7$, $t_{12} = 5$ days, $t_{22} = 15$ days

For time Period 3,

$F_3 = 0.0$, $t_{13} = 7$ days, $t_{23} = 8$ days

For time Period 4,

$F_4 = 0.7$, $t_{14} = 8$ days, $t_{24} = 0$ days

The equation for Nuclide i would be:

$$\begin{aligned} PF_i = {} & [1.0\,(1-e^{-\lambda_i\,(10\text{ days})})e^{-\lambda_i\,(20\text{ days})}] \\ & + [0.7\,(1-e^{-\lambda_i\,(5\text{ days})})e^{-\lambda_i\,(15\text{ days})}] \\ & + [0.0\,(1-e^{-\lambda_i\,(7\text{ days})})e^{-\lambda_i\,(8\text{ days})}] \\ & + [0.7\,(1-e^{-\lambda_i\,(8\text{ days})})e^{-\lambda_i\,(0\text{ days})}] \end{aligned}$$

TITLE: SOURCE INVENTORY POWER CORRECTION FOR VARIABLE POWER HISTORY

ATTACHMENT 4 (Continued)

CORRECTION FOR VARIABLE POWER LEVEL

| | A | B | C | D |
|---|---|---|---|---|
| NUCLIDE | Decay Constant ($day^{-1}$) | Power Correction Factor ($PF_i$) | Equilibrium Source Inventory (Ci) | Corrected Source Inventory (Ci) |
| Gap Inventory | | | | |
| Kr-87 | 13.10 | | 3.9E+4 | |
| Xe-133 | 0.13 | | 1.3E+6 | |
| I-131 | $8.62X10^{-2}$ | | 8.0E+5 | |
| I-132 | 6.94 | | 1.3E+5 | |
| Fuel Pellet Inventory | | | | |
| Kr-87 | 13.10 | | 5.9E+7 | |
| Xe-133 | 0.13 | | 1.9E+8 | |
| Te-132 | $2.13X10^{-1}$ | | 1.4E+8 | |
| Cs-134 | $9.21X10^{-4}$ | | 3.1E+6 | |
| Ba-140 | $5.42X10^{-2}$ | | 1.6E+8 | |
| La-140 | $4.15X10^{-1}$ | | 1.8E+E | |
| I-131 | $8.62X10^{-2}$ | | 9.7E+7 | |
| I-132 | 6.94 | | 1.4E+8 | |

ATTACHMENT 5

1. Copy both the total waterborne and airborne nuclide inventories into Columns 1 and 2 of the attached Form.

2. Add Columns 1 and 2 to get the total release inventory of each nuclide of interest, and record the result in Column 3.

3. Copy the expected source inventory into Column 4 from Attachment 3 or 4.

4. For each nuclide, divide the value in Column 3 by the corresponding value in Column 4, and record the result in Column 5.

5. Go to Section 2.2 Step 5.

TITLE: COMPARISON OF EXPECTED AND ACTUAL SOURCE INVENTORIES

ATTACHMENT 5

| | 1 | 2 | 3 | 4 | 5 |
|---|---|---|---|---|---|
| Isotope | Total Waterborne (Attachment 1) | Total Airborne (Attachment 2) | Total Released Inventory (Ci) | Expected Source Inventory (Ci) (Attachment 3 or 4) | Percent Released |
| Gas Gap Inventory | | | | | |
| Kr-87 | | | | | |
| Xe-133 | | | | | |
| I-131 | | | | | |
| I-132 | | | | | |
| | | | | Average % | ______ |
| Fuel Pellet Inventory | | | | | |
| Kr-87 | | | | | |
| Xe-133 | | | | | |
| I-131 | | | | | * |
| I-132 | | | | | * |
| Cs-134 | | | | | * |
| Te-132 | | | | | * |
| Ba-140 | | | | | |
| La-140 | | | | | |
| | | | | Average % | |

*Do not use for calculating "average percent". Use only for qualitative assessment on Attachment 6.

Check One:
___ No damage
___ GAP release
___ Core melt

ATTACHMENT 6

1. For each nuclide listed on the attached worksheet, check the box which corresponds to the inventory percentage found on attachment 5. Circle the applicable iodine ratio.

2. To best determine the category of damage, concentrate on the presence or absence of key nuclides (e.g., Te, Cs, Ba, La).

3. The general location of the marks should give an indication of the type of core damage.

4. Enter the type of damage that has been determined on Attachment 5, and proceed to step 2.2.6.

   NOTE: If Ag-110m was found in any of the samples, it is a good indicator of fuel melt (Ag is from the control rods).

## ATTACHMENT 6 (Continued)
## GRAPHIC DAMAGE ASSESSMENT WORKSHEET

| | No Damage | GAP Release | | | Fuel Overheat | | | Fuel Melt | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | <0.12% | <10%* | 10-50% | >50% | <10%* | 10-50% | >50% | <10%* | 10-50% | >50% |
| Kr-87 | | | | | | | | | | |
| Xe-133 | | | | | | | | | | |
| I-131 | | | | | | | | | | |
| I-132 | | | | | | | | | | |
| Iodine Ratio | | $\frac{I\text{-}132}{I\text{-}131} < 0.2$ | | | $0.2 \leq \frac{I\text{-}132}{I\text{-}131} \leq 1.5$ | | | $\frac{I\text{-}132}{I\text{-}131} \geq 1.5$ | | |
| Cs-134 | | | | | | | | | | |
| Te-132 | | | | | | | | | | |
| Ba-140 | | | | | | | | | | |
| La-140 | | | | | | | | | | |

*Do not check if below 0.12%.