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NUCLEAR OPERATIONS DEPARTMENT

PILGRIM NUCLEAR POWER STATION

Procedure No. 5.7.5

ESTIMATING CORE DAMAGE

List of Effective Pages

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Approved

Charles White
ORC Chairman

Date

March 6, 1983

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I. PURPOSE

The purpose of this procedure is to provide a mechanism to determine the degree of Reactor Core Damage from samples collected using the Post Accident Sampling System (PASS) and other available plant parameters.

II. DISCUSSION

Two procedures have been developed by G.E. which provide the user with a rapid mechanism to preliminarily identify what (if any) core damage has occurred during and after a suspected degraded core event. This procedure (Attachment C) has been developed to predict the degree of core damage based on fission product concentrations in either the water or gas samples. These concentrations are determined from samples collected by the PASS System (Reference Procedures NO. 5.7.4.1.0 through 5.7.4.1.4) and analyzed in the Radio Chem. Lab. as described in Procedure No.'s 7.11.6 through 7.11.9.

The second procedure (Attachment D) supplements the above by providing the user with a mechanism to verify and refine the initial estimates of core damage. This verification is based on correlations established between the radioisotopic information and other significant plant parameters available during and after the event.

III. REFERENCES

- A. Emergency Procedures 5.7.4.1, 5.7.4.1.1 through 5.7.4.1.14
- B. Chemical Analytical Procedures 7.11.6 through 7.11.9
- C. Operating Procedures 2.2.133
- D. BWR Owners Group Letter BWROG-8324 dated June 17, 1983

IV. PREREQUISITES

- A. Samples collected and analyzed as directed by the Watch Engineer.

V. APPARATUS

None

VI. PRECAUTIONS

None

VII. LIMITATIONS

None

VIII. PROCEDURE

- A. Assess the degree of core damage as described in Attachment A.
- B. Verify the assessment by integrating other plant parameters as described in Attachment B.

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ATTACHMENT A

- I. Procedure to determine Core Damage based on samples collected from the Jet Pump, RHR and Torus Sample Points.

- A. Set CW_{I-131} and CW_{CS-137} as the measured concentrations (in uCi/g) of I-131 and CS-137 from the Jet Pump Sample Points.

$$CW_{I-131} = \underline{\hspace{2cm}}$$

$$CW_{CS-137} = \underline{\hspace{2cm}}$$

- B. Correct the measured concentrations of the above radionuclides to the time of reactor shutdown as follows:

1. Let; CW_{I-131}^{JP} and CW_{CS-137}^{JP} = Concentration of the radionuclide of interest at the time of reactor shutdown.

t = Time period (hrs.) between reactor shutdown and the determination of the radioactivity of the radionuclide of interest

2. Find CW_{I-131}^{JP} where

$$CW_{I-131}^{JP} = CW_{I-131} \times e^{3.5914 \text{ E-}03 \times t}$$

$$CW_{I-131}^{JP} = \underline{\hspace{2cm}}$$

3. Find CW_{CS-137}^{JP} where

$$CW_{CS-137}^{JP} = CW_{CS-137} \times e^{2.6221 \text{ E-}06 \times t}$$

$$CW_{CS-137}^{JP} = \underline{\hspace{2cm}}$$

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- C. Set C_{WI-131} and $C_{WCS-137}$ as the measured concentrations (in $\mu\text{Ci/g}$) of I-131 and CS-137 from the RHR Sample Points.

$C_{WI-131} = \underline{\hspace{2cm}}$

$C_{WCS-137} = \underline{\hspace{2cm}}$

- D. Correct the measured concentrations of the above radionuclides to the time of reactor shutdown as follows:

1. Let; C_{WI-131}^{RHR} and $C_{WCS-137}^{\text{RHR}}$ = Concentration of the radionuclide of interest at the time of reactor shutdown.

t = Time period (hrs.) between reactor shutdown and the determination of the radioactivity of the radionuclide of interest

2. Find C_{WI-131}^{RHR} where

$$C_{WI-131}^{\text{RHR}} = C_{WI-131} \times e^{3.5914 \text{ E-03} \times t}$$

$$C_{WI-131}^{\text{RHR}} = \underline{\hspace{2cm}}$$

3. Find $C_{WCS-137}^{\text{RHR}}$ where

$$C_{WCS-137}^{\text{RHR}} = C_{WCS-137} \times e^{2.6221 \text{ E-06} \times t}$$

$$C_{WCS-137}^{\text{RHR}} = \underline{\hspace{2cm}}$$

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- E. Set C_{WI-131} and $C_{WCS-137}$ as the measured concentrations (in $\mu\text{Ci/g}$) of I-131 and CS-137 from the Torus Sample Points.

$$C_{WI-131} = \underline{\hspace{2cm}}$$

$$C_{WCS-137} = \underline{\hspace{2cm}}$$

- F. Correct the measured concentrations of the above radionuclides to the time of reactor shutdown as follows:

1. Let; C_{WI-131}^T and $C_{WCS-137}^T$ = Concentration of the radionuclide of interest at the time of reactor shutdown.

t = Time period (hrs.) between reactor shutdown and the determination of the radioactivity of the radionuclide of interest

2. Find C_{WI-131}^T where

$$C_{WI-131}^T = C_{WI-131} \times e^{3.5914 \text{ E-}03 \times t}$$

$$C_{WI-131}^T = \underline{\hspace{2cm}}$$

3. Find $C_{WCS-137}^T$ where

$$C_{WCS-137}^T = C_{WCS-137} \times e^{2.6221 \text{ E-}06 \times t}$$

$$C_{WCS-137}^T = \underline{\hspace{2cm}}$$

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6. Determine average fission product concentration of the nuclide of interest in the total coolant mass as follows:

1. Let C_{wia} : Average fission product concentration in the plant coolant mass for the isotope of interest.

- a. Find $C_{wiaI-131}$ as follows:

$$C_{wiaI-131} = \frac{C_{wiI-131}(\text{Step B. or D.}) \times 2.05 \text{ E } 08 + C_{wiI-131}(\text{Step F}) \times 2.38 \text{ E } 09}{2.585 \text{ E } 09}$$

$$C_{wiaI-131} = \underline{\hspace{2cm}}$$

- b. Find $C_{wiaCS-137}$ as follows:

$$C_{wiaCS-137} = \frac{C_{wiCS-137}(\text{Step B. or D.}) \times 2.05 \text{ E } 08 + C_{wiCS-137}(\text{Step F}) \times 2.38 \text{ E } 09}{2.585 \text{ E } 09}$$

$$C_{wiaCS-137} = \underline{\hspace{2cm}}$$

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H. Calculate the fission product inventory factor as follows:

1. Let FI_{I-131} and FI_{CS-137} = Fission Product inventory correction factor for the radionuclide of interest at the time of reactor shutdown.
- P_j = Steady state reactor power level operated during period number j in MWt - Note the variation in power level should be limited to $\pm 20\%$ to meet steady state conditions.
- j = NUMBER of operating periods at a steady state power level since startup from the last refueling outage to the last shutdown prior to sampling.
- T_j = Duration of operation @ steady state power for the operating period j in HOURS.
- T_j^0 = Time between the end of the " j th" operating period and the last reactor shutdown prior to sampling in HOURS.

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2. Find FI_{I-131} where FI_{I-131} =

$$\frac{3.651 \text{ E } 03}{\sum_j [P_j (1 - e^{-3.5914 \text{ E } -03 \times T_j})(e^{-3.5914 \text{ E } -03 \times T_j})]}$$

$$FI_{I-131} = \underline{\hspace{2cm}}$$

3. Find FI_{CS-137} where FI_{CS-137} =

$$\frac{2.431 \text{ E } 02}{\sum_j [P_j (1 - e^{-2.6221 \text{ E } -06 \times T_j})(e^{-2.6221 \text{ E } -06 \times T_j})]}$$

$$FI_{CS-137} = \underline{\hspace{2cm}}$$

- I. Calculate the normalized concentration of the radioisotope of interest as follows:

1. Let Cw_{I-131}^{REF} & Cw_{CS-137}^{REF} = Normalized concentration of the radioisotope of interest at the time of reactor shutdown.

2. Find Cw_{I-131}^{REF} where Cw_{I-131}^{REF} =

$$Cw_{I-131}^{REF} \times FI_{I-131} \times 6.594 \text{ E-01}$$

$$Cw_{I-131}^{REF} = \underline{\hspace{2cm}}$$

3. Find Cw_{CS-137}^{REF} where Cw_{CS-137}^{REF} =

$$Cw_{CS-137}^{REF} \times FI_{CS-137} \times 6.594 \text{ E-01}$$

$$Cw_{CS-137}^{REF} = \underline{\hspace{2cm}}$$

- J. Refer to Figures 1 and 2 to estimate the extent of fuel or cladding damages.

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II. Procedure to determine Core Damage based on samples collected from the Torus and Drywell Atmosphere Sample Points.

- A. Set Cg_{Xe-133} and Cg_{Kr-85} as the measured concentrations (in uCi/g) of Xe-133 and Kr-85 from the Drywell Sample Points.

Cg_{Xe-133} = _____

Cg_{Kr-85} = _____

- B. Correct the measured concentrations of the above radionuclides to the time of reactor shutdown as follows:

1. Let; Cg_{Xe-133}^D and Cg_{Kr-85}^D = Concentration of the radionuclide of interest at the time of reactor shutdown.
 t = Time period (hrs.) between reactor shutdown and the determination of the radioactivity of the radionuclide of interest

2. Find Cg_{Xe-133}^D where

$$Cg_{Xe-133}^D = Cg_{Xe-133} \times e^{5.5 \text{ E-}03 \times t}$$

$$Cg_{Xe-133}^D = \underline{\hspace{2cm}}$$

3. Find Cg_{Kr-85}^D where

$$Cg_{Kr-85}^D = Cg_{Kr-85} \times e^{7.3796 \text{ E-}06 \times t}$$

$$Cg_{Kr-85}^D = \underline{\hspace{2cm}}$$

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- C. Correct the measured atmospheric samples for temperature and pressure difference between the sample vial and Drywell.

1. Let $C_{gicKr-85}^D$ and $C_{gicXe-133}^D$ = Concentration of the radionuclides of interest @ the Drywell atmospheric temperature and pressure @ the time of reactor shutdown.

T_1 = Temperature of the gas in the sample vial. °F

P_1 = Pressure of the gas in the sample vial. PSIA

T_2 = Temperature of the gas in the containment. °F

P_2 = Pressure of the gas in the containment. PSIA

2. Find $C_{gicKr-85}^D$ where:

$$C_{gicKr-85}^D = C_{giKr-85}^D \frac{P_2 T_1}{P_1 T_2}$$

$$C_{gicKr-85}^D = \underline{\hspace{2cm}}$$

3. Find $C_{gicXe-133}^D$ where:

$$C_{gicXe-133}^D = C_{giXe-133}^D \frac{P_2 T_1}{P_1 T_2}$$

$$C_{gicXe-133}^D = \underline{\hspace{2cm}}$$

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- D. Set Cg_{Xe-133} and Cg_{Kr-85} as the measured concentrations (in $\mu Ci/g$) of Xe-133 and Kr-85 from the Torus Sample Point.

$$Cg_{Xe-133} = \underline{\hspace{2cm}}$$

$$Cg_{Kr-85} = \underline{\hspace{2cm}}$$

- E. Correct the measured concentrations of the above radionuclides to the time of reactor shutdown as follows:

1. Let; Cg_{Xe-133}^T and Cg_{Kr-85}^T = Concentration of the radionuclide of interest at the time of reactor shutdown.

t = Time period (hrs.) between reactor shutdown and the determination of the radioactivity of the radionuclide of interest

2. Find Cg_{Xe-133}^T where

$$Cg_{Xe-133}^T = Cg_{Xe-133} \times e^{5.5 \times 10^{-3} \times t}$$

$$Cg_{Xe-133}^T = \underline{\hspace{2cm}}$$

3. Find Cg_{Kr-85}^T where

$$Cg_{Kr-85}^T = Cg_{Kr-85} \times e^{7.3796 \times 10^{-6} \times t}$$

$$Cg_{Kr-85}^T = \underline{\hspace{2cm}}$$

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- F. Correct the measured atmospheric samples for temperature and pressure difference between the sample vial and Torus.

1. Let $C_{gicKr-85}^T$ and $C_{gicXe-133}^T$ = Concentration of the radionuclides of interest @ the Torus atmospheric temperature and pressure @ the time of reactor shutdown.
- T_1 = Temperature of the gas in the sample vial. °F
- P_1 = Pressure of the gas in the sample vial. PSIA
- T_2 = Temperature of the gas in the containment. °F
- P_2 = Pressure of the gas in the containment. PSIA

2. Find $C_{gicKr-85}^T$ where:

$$C_{gicKr-85}^T = C_{giKr-85}^T \frac{P_2 T_1}{P_1 T_2}$$

$$C_{gicKr-85}^T = \underline{\hspace{2cm}}$$

3. Find $C_{gicXe-133}^T$ where:

$$C_{gicXe-133}^T = C_{giXe-133}^T \frac{P_2 T_1}{P_1 T_2}$$

$$C_{gicXe-133}^T = \underline{\hspace{2cm}}$$

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6. Determine average fission product concentration of the nuclide of interest in the total containment volume as follows:

1. Let C_{gia} : Average fission product concentration in the containment volume for the isotope of interest.

- a. Find $C_{giaXe-133}$ as follows:

$$C_{giaXe-133} = \frac{(C_{gicD_{Xe-133}} \times 4.16 \text{ E } 09) + (C_{gicT_{Xe-133}} \times 3.18 \text{ E } 09)}{7.34 \text{ E } 09}$$

$$C_{giaXe-133} = \underline{\hspace{2cm}}$$

- b. Find $C_{giaKr-85}$ as follows:

$$C_{giaKr-85} = \frac{(C_{gicD_{Kr-85}} \times 4.16 \text{ E } 09) + (C_{gicT_{Kr-85}} \times 3.18 \text{ E } 09)}{7.34 \text{ E } 09}$$

$$C_{giaKr-85} = \underline{\hspace{2cm}}$$

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H. Calculate the fission product inventory factor as follows:

1. Let FII_{Xe-133} and FII_{Kr-85} = Fission Product inventory correction factor for the radionuclide of interest at the time of reactor shutdown.

P_i = Steady state reactor power level operated during period number j in MWt - Note the variation in power level should be limited to $\pm 20\%$ to meet steady state conditions.

j = NUMBER of operating periods at a steady state power level since startup from the last refueling outage to the last shutdown prior to sampling.

T_j = Duration of operation @ steady state power for the operating period j in HOURS.

T_j^o = Time between the end of the "jth" operating period and the last reactor shutdown prior to sampling in HOURS.

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2. Find FII_{Kr-85} where $FII_{Kr-85} =$

$$\frac{6.436 \text{ E } 02}{\sum_j [P_j (1 - e^{-7.3796 \text{ E } -06 \times T_j})(e^{-7.3796 \text{ E } -06 \times T_j^o})]}$$

$FII_{Kr-85} =$ _____

3. Find FII_{Xe-133} where $FII_{Xe-133} =$

$$\frac{3.651 \text{ E } 03}{\sum_j [P_j (1 - e^{-5.5 \text{ E } -03 \times T_j})(e^{-5.5 \text{ E } -03 \times T_j^o})]}$$

$FII_{Xe-133} =$ _____

- I. Calculate the normalized concentration of the radioisotope of interest as follows:

1. Let C_{gKr-85}^{REF} & $C_{gXe-133}^{REF}$ = Normalized concentration of the radioisotope of interest at the time of reactor shutdown.

2. Find C_{gKr-85}^{REF} where C_{gKr-85}^{REF} =

$$C_{gKr-85}^{REF} \times F_{iKr-85} \times 1.835 \text{ E-01}$$

$$C_{gKr-85}^{REF} = \underline{\hspace{2cm}}$$

3. Find $C_{gXe-133}^{REF}$ where $C_{gXe-133}^{REF}$ =

$$C_{gXe-133}^{REF} \times F_{iXe-133} \times 1.835 \text{ E-01}$$

$$C_{gXe-133}^{REF} = \underline{\hspace{2cm}}$$

- J. Refer to Figures 3 and 4 to estimate the extent of fuel or cladding damages.

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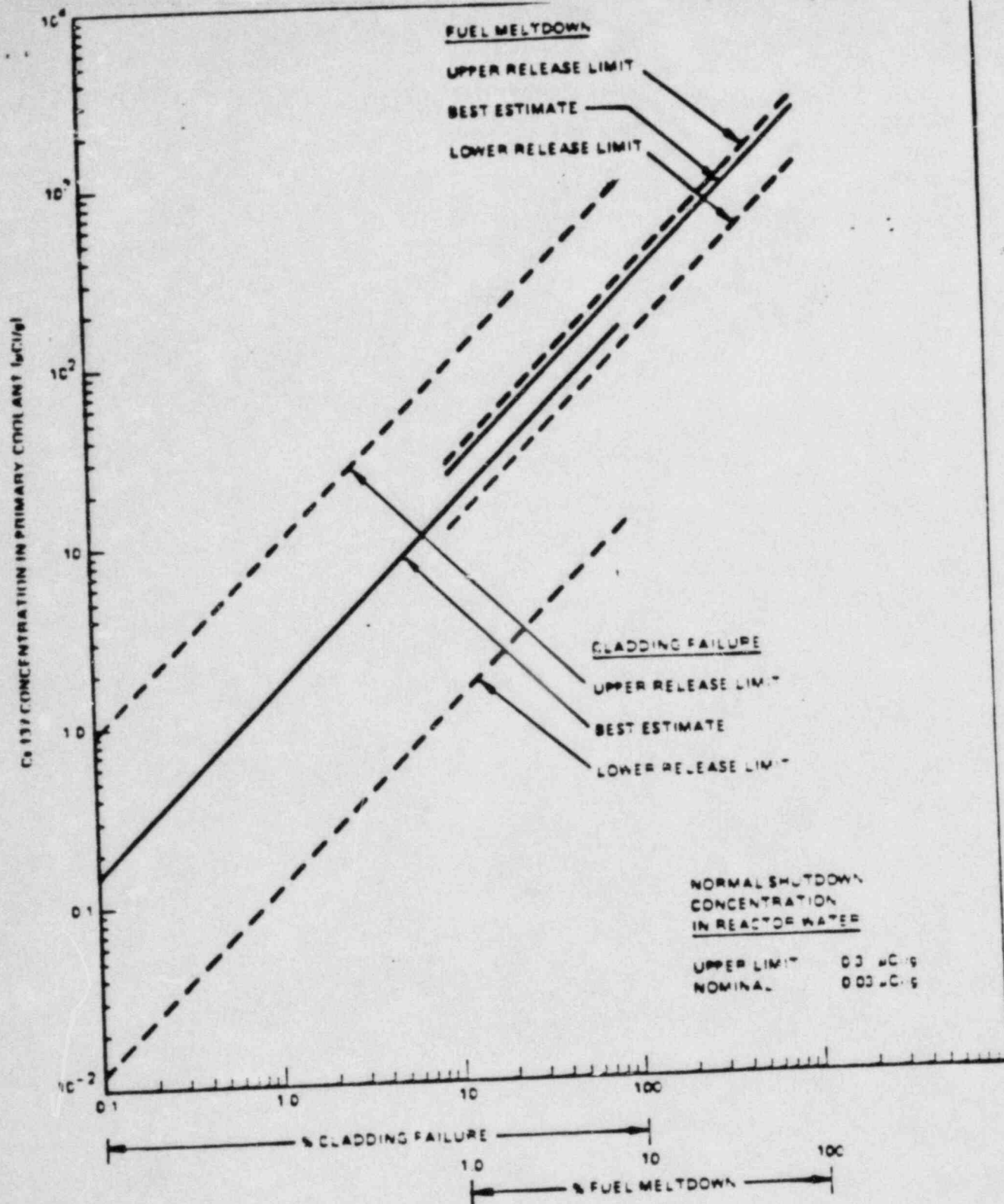


Figure 2. Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant.

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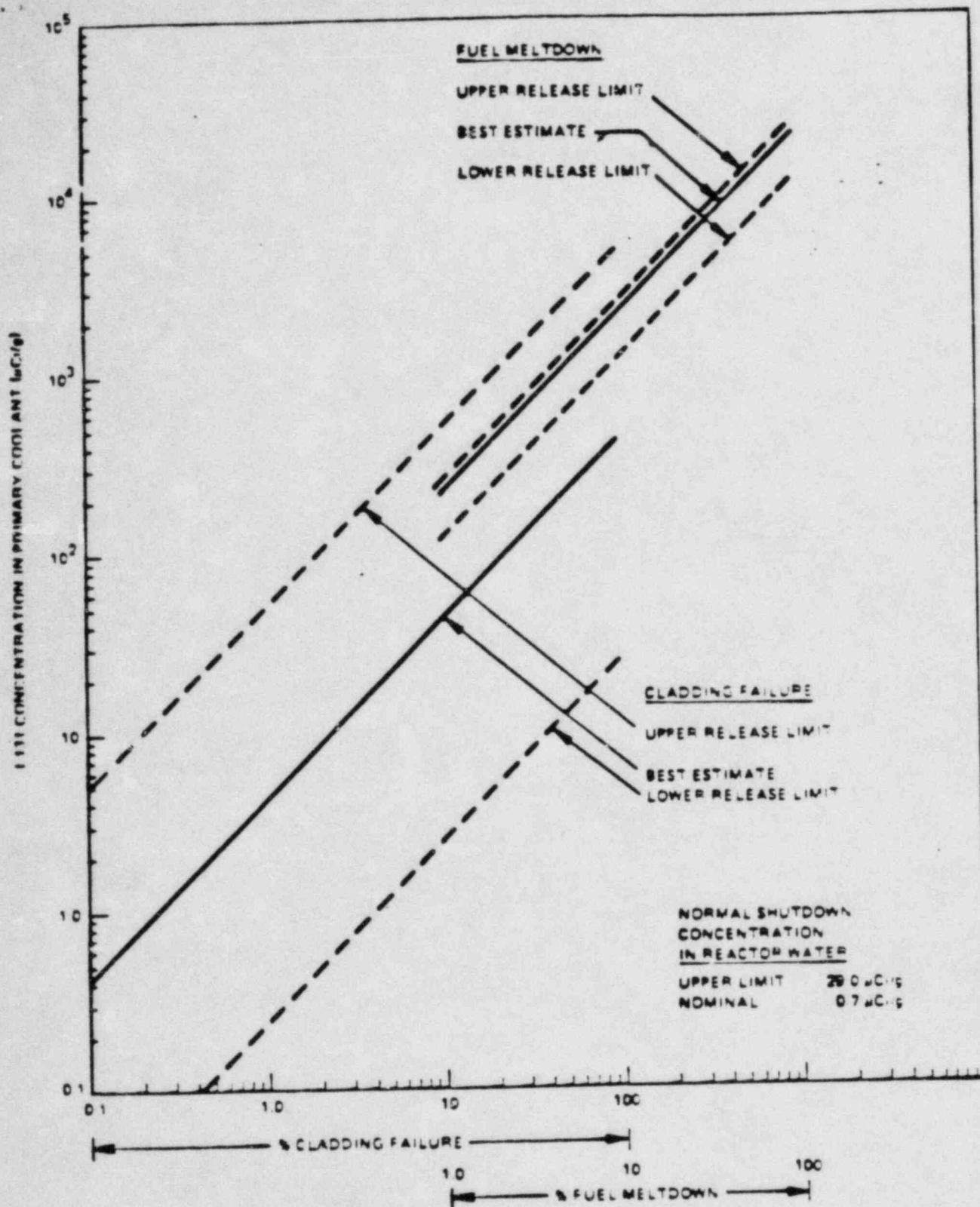


Figure 1. Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant.

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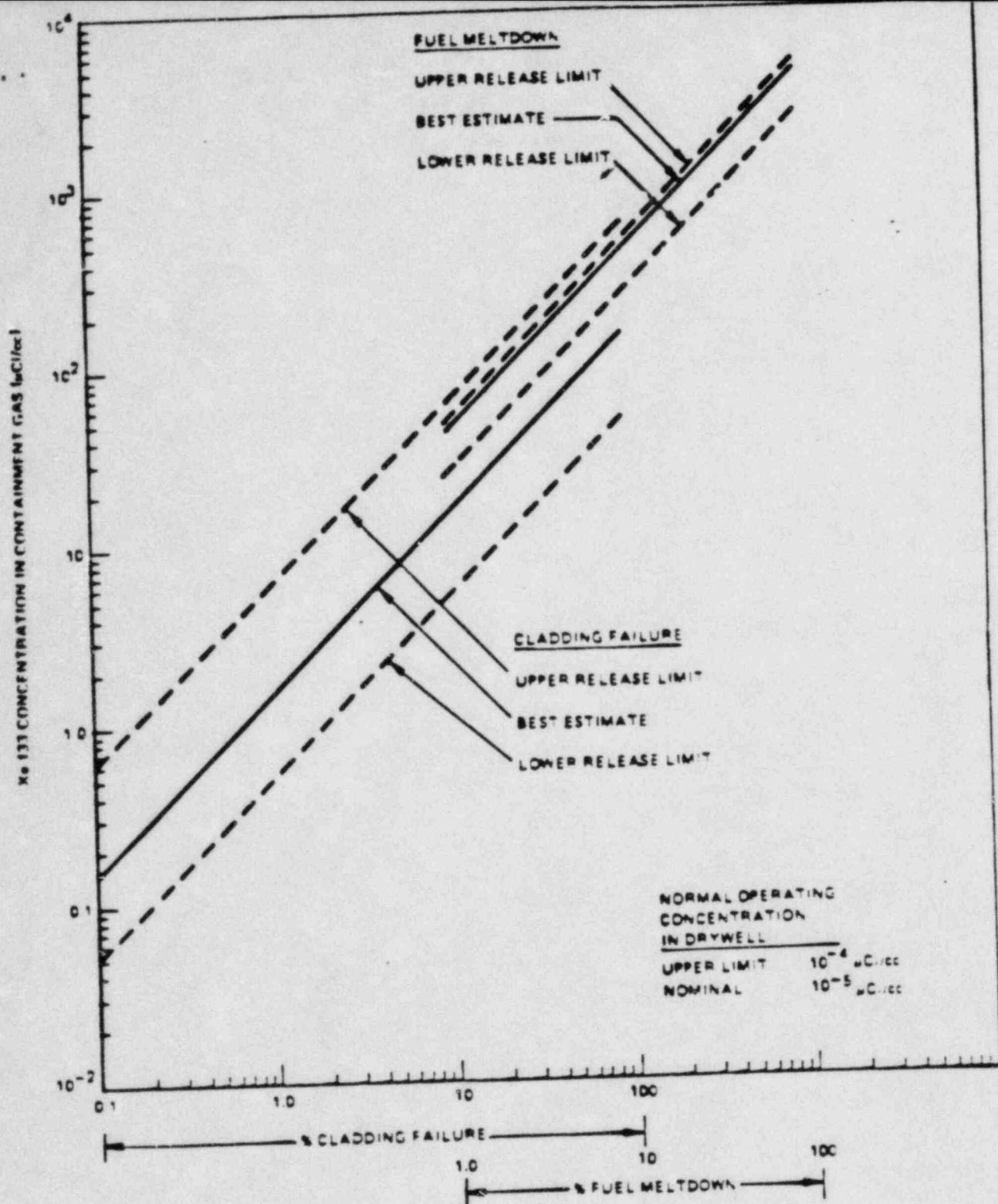


Figure 3. Relationship Between Xe-133 Concentration in the Containment Gas (Drywell + Torus Gas) and the Extent of Core Damage in Reference Plant.

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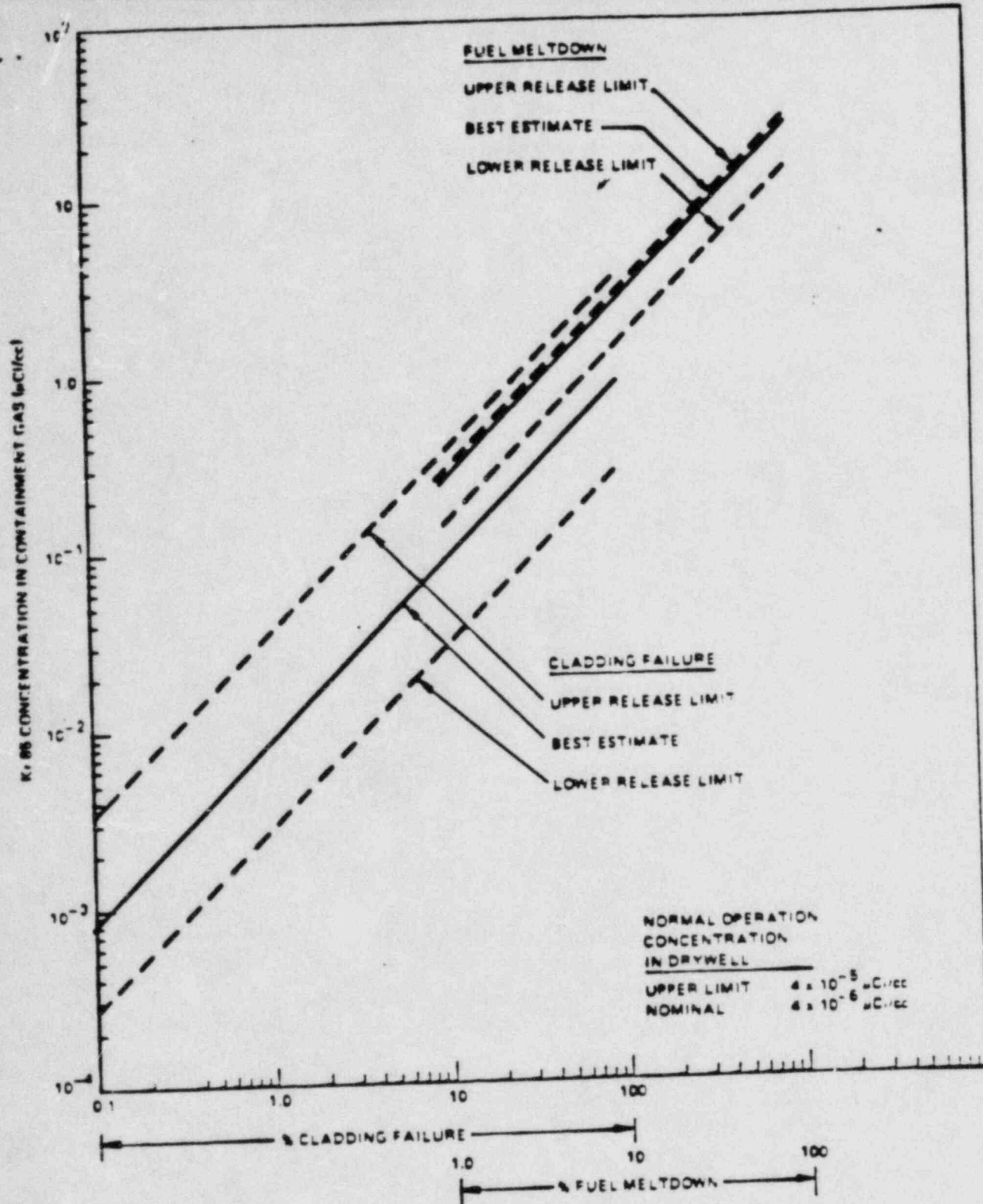


Figure 4. Relationship Between Kr-85 Concentration in the Containment Gas (Drywell + Torus Gas) and the Extent of Core Damage in Reference Plant.

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ATTACHMENT B

- I. Procedure to determine the degree of cladding damage from analysis of the torus atmosphere H₂ monitor (in panel C-174 or C-175 indicators AIT-2A-5082A or AIT-2A-5082B respectively) readings.

- A. Set H_D as the measured concentration (%) of hydrogen in the torus as indicated on the torus atmosphere monitors (panel C-174 or C-175)

H_D = _____

- B. From Figure 1 determine the metal water reaction (%) for the reference plant.

MW_{REF} = _____

- C. Find the amount of cladding that has reacted (therefore is damaged) to form hydrogen as follows:

1. Let %MW = The percentage of the cladding that has reacted to form hydrogen.
2. Find %MW where %MW = MW_{REF} x 6.3845 E-01

%MW = _____

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II. Procedure to determine the degree of cladding damage from analysis of the drywell atmosphere H_2 monitor (in panel C-174 or C-175 indicators AIT-2A-5082A or AIT-2A-5082B respectively) readings.

- A. Set H_D as the measured concentration (%) of hydrogen in the torus as indicated on the torus atmosphere monitors (panel C-174 or C-175)

$$H_D = \underline{\hspace{2cm}}$$

- B. From Figure 1 determine the metal water reaction (%) for the reference plant.

$$MW_{REF} = \underline{\hspace{2cm}}$$

- C. Find the amount of cladding that has reacted (therefore is damaged) to form hydrogen as follows:

1. Let $\%MW$ = The percentage of the cladding that has reacted to form hydrogen.
2. Find $\%MW$ where $\%MW = MW_{REF} \times 6.3845 \text{ E-01}$

$$\%MW = \underline{\hspace{2cm}}$$

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- III. Procedure to determine the degree of Fuel Inventory released from analysis of the Containment High Range Monitoring System (CHRMS).

NOTE

A Drywell CHRM and a Torus CHRM should be used together to determine the degree of Fuel Inventory released.

- A. Obtain the associated CHRM reading.

Ch. A / B

Drywell, $R_D = \frac{\quad}{\quad}$ R/hr.

Torus, $R_T = \frac{\quad}{\quad}$ R/hr.

- B. Record the time, in hours, after reactor scram that readings were taken.

$t = \frac{\quad}{\quad}$ hrs

- C. From Figure 2 determine the following:

	<u>DRYWELL</u>	<u>TORUS</u>
	CH. A / B	CH. A / B
1. Fuel Failure Type:	<u> </u> / <u> </u>	<u> </u> / <u> </u>
2. % of Noble Gas:	<u> </u> / <u> </u>	<u> </u> / <u> </u>
3. % of Halogens:	<u> </u> / <u> </u>	<u> </u> / <u> </u>
4. % of Solids:	<u> </u> / <u> </u>	<u> </u> / <u> </u>

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HYDROGEN (dry basis) MARK 1/11: 380,000 FT³/500 BUNDLES

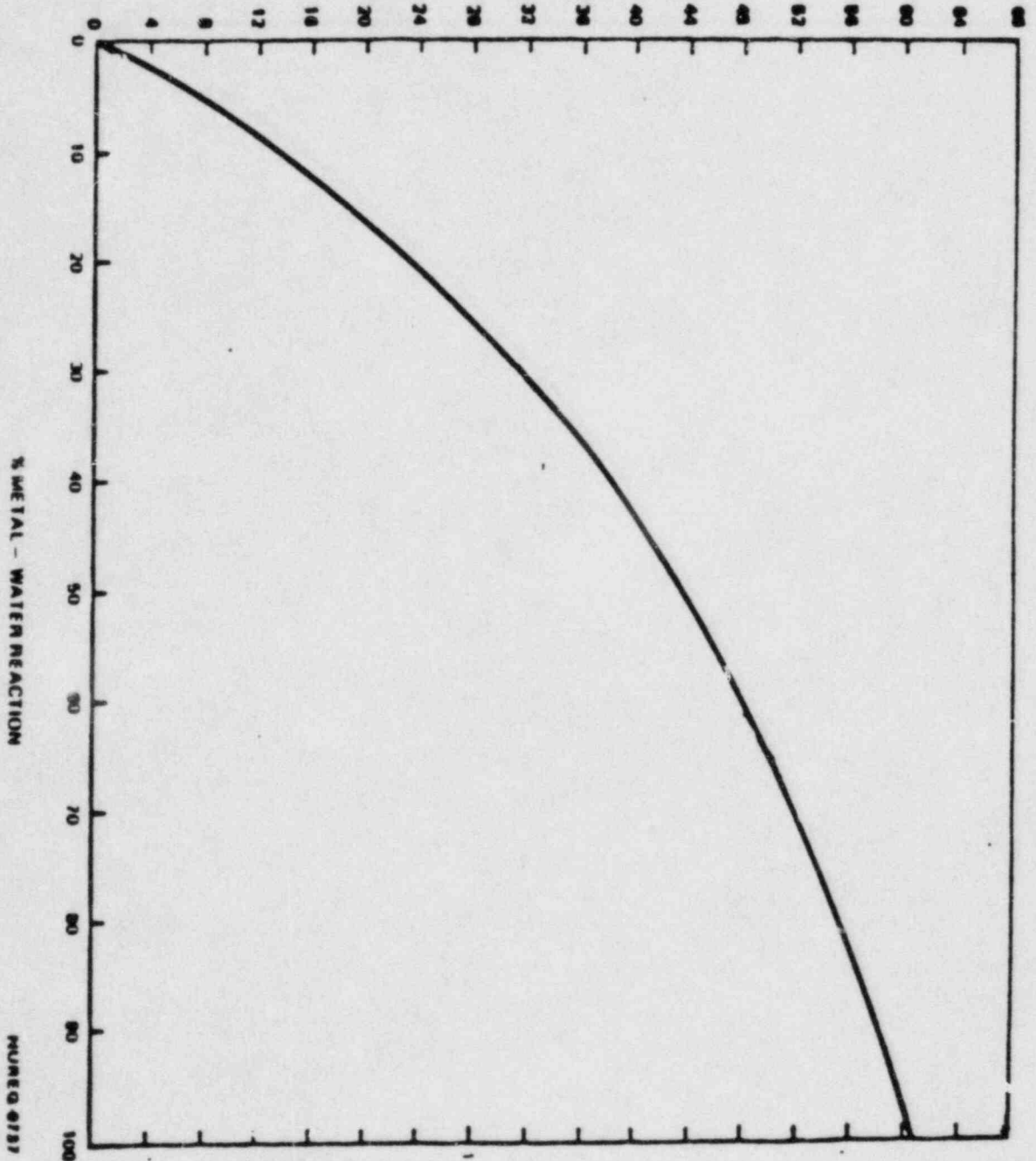
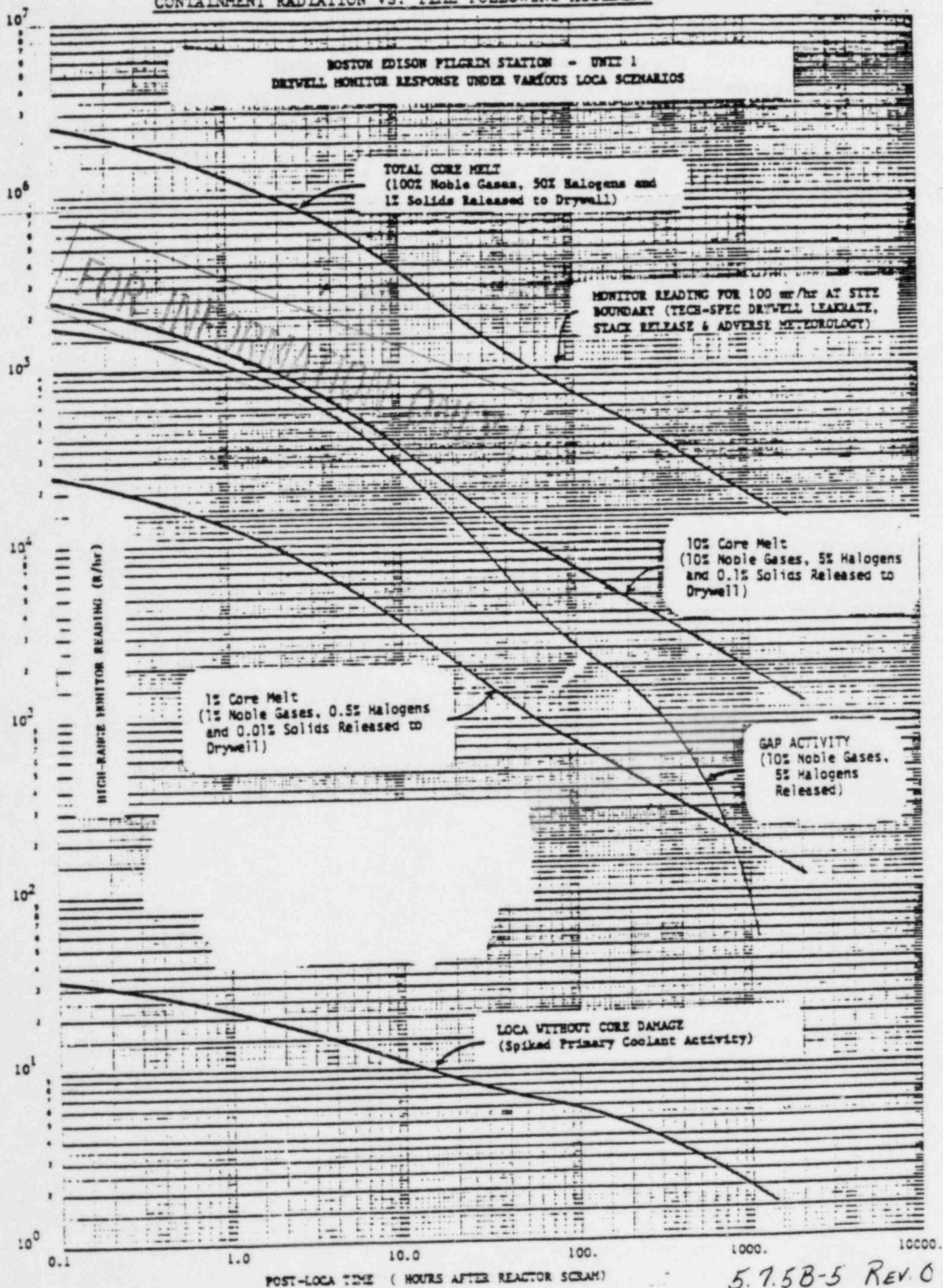


Figure 1. Hydrogen Concentration for Mark 1/11 as a Function of Metal-Water Reaction

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FIGURE 2
CONTAINMENT RADIATION VS. TIME FOLLOWING ACCIDENT



ATTACHMENT 1

PROCEDURES FOR THE DETERMINATION
OF THE EXTENT OF CORE DAMAGE
UNDER ACCIDENT CONDITIONS

PROCEDURES FOR THE DETERMINATION OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

C. C. LIN

GENERAL  ELEC

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Class I
August 1982

PROCEDURES FOR THE DETERMINATION OF THE
EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

Chien C. Lin

Approved: _____

R L Cowan
R. L. Cowan, Manager
Plant Chemical and Radiation Technology

Approved: _____

E Kiss
E. Kiss, Manager
Plant Technology

NUCLEAR POWER SYSTEM ENGINEERING DEPARTMENT, GENERAL ELECTRIC COMPANY
VALLECITOS NUCLEAR CENTER, PLEASANTON, CA 94566

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1. OBJECTIVE AND SCOPE

The purpose of this procedure is to determine the degree of reactor core damage from the measured fission product concentrations in either the water or gas samples taken from the primary system under accident conditions. The procedure involves calculations of fission product inventories in the core and the release of inventories into the primary system under postulated loss-of-coolant accident (LOCA) conditions. The fuel gap fission products are assumed to be released upon the rupture of fuel cladding. The majority of fission product inventories in the fuel rods would be released when the fuel is melted at higher temperatures. A BWR-6/238 with a Mark III containment is used as a reference plant in the demonstration of this procedure. Application of the procedure for any other type or size of boiling water reactor (BWR) is described.

2. PROCEDURES FOR DETERMINATIONS OF CORE DAMAGE

2.1 REFERENCE PLANT (BWR-6/238, MARK III) ✓

2.1.1 Reference Plant Parameters

The pertinent plant parameters for the reference plant are given below:

Rated reactor thermal power	3579 MWt
Number of fuel bundles	748 bundles
Total primary coolant mass (reactor water plus suppression pool water)	3.92×10^9 g
Total containment and drywell gas space volume	4.0×10^{10} cc

The fission product inventories in the core are calculated based on three years (1095 days) of continuous operation at 3651 MWt, or 102% of rated power, by using a computer code developed at Los Alamos and adapted to the GE computer system.¹ The inventories of some major fission products in the core at the time of reactor shutdown are given in Table 1.

2.1.2 Procedure

Either the gas or water samples taken from the post accident sampling system are analyzed for major fission product concentrations by gamma ray spectrometry. If the concentration of a fission product in reactor water or drywell, corrected the decay to the time of reactor shutdown, is measured to be higher than the baseline concentration shown in Table 2 (see Section 3.1 for details), the extent of fuel or cladding damage can be determined directly from Figures 1 through 4 based on isotopes I-131, Cs-137, Xe-133, and Kr-85. Measurements of Cs-137 and Kr-85 activities are not very likely until the reactor has been shut down for longer than a few weeks and most of the shorter-lived isotopes have decayed.

Table 1

CORE INVENTORY OF MAJOR FISSION PRODUCTS IN A
REFERENCE PLANT OPERATED AT 3651 MWt FOR THREE YEARS

Chemical Group	Isotope*	Half-Life	Inventory** 10 ⁶ Ci	Major Gamma Ray Energy (Intensity) KeV (γ/d)
Noble gases	Kr-35m	4.48h	24.6	151(0.753)
	Kr-85	10.72y	1.1	514(0.0044)
	Kr-87	76.3m	47.1	403(0.495)
	Kr-88	2.84h	66.8	196(0.26), 1530(0.109)
	Xe-133	5.25d	202.0	81(0.365)
	Xe-135	9.11h	26.1	250(0.899)
Halogens	I-131	8.04d	96.0	364(0.812)
	I-132	2.3h	140	668(0.99), 773(0.762)
	I-133	20.8h	201	530(0.86)
	I-134	52.6m	221	847(0.954), 884(0.653)
	I-135	6.61h	189	1132(0.225), 1260(0.286)
Alkali Metals	Cs-134	2.06y	19.6	605(0.98), 796(0.85)
	Cs-137	30.17y	12.1	662(0.85)
	Cs-138	32.2m	178.0	463(0.307), 1436(0.76)
Tellurium Group	Te-132	78.2h	138	228(0.88)
Noble Metals	Mo-99	66.02h	183	740(0.128)
	Ru-103	39.4d	155	497(0.89)
Alkaline Earths	Sr-91	9.5h	115	750(0.23), 1024(0.325)
	Sr-92	2.71h	123	1388(0.9)
	Ba-140	12.8d	173	537(0.254)
Rare Earths	Y-92	3.54h	124	934(0.139)
	La-140	40.2h	184	487(0.455), 1597(0.955)
	Ce-141	32.5d	161	145(0.48)
	Ce-144	284.3d	129	134(0.108)
Refractories	Zr-95	64.0d	161	724(0.437), 757(0.553)
	Zr-97	16.9h	166	743(0.928)

*Only the representative isotopes which have relatively large inventory and considered to be easy to measure are listed here.

**At the time of reactor shutdown.

Table 2

FISSION PRODUCT CONCENTRATIONS IN REACTOR WATER
AND DRYWELL GAS SPACE DURING REACTOR SHUTDOWN UNDER NORMAL CONDITIONS

Isotope	Reactor Water, $\mu\text{Ci/g}$		Drywell Gas ($\mu\text{Ci/cc}$)	
	Upper Limit	Nominal	Upper Limit	Nominal
I-131	29	0.7	---	---
Cs-137 ^c	0.3 ^a	0.03 ^b	---	---
Xe-133	---	---	10^{-4a}	10^{-5b}
Kr-85	---	---	4×10^{-5a}	4×10^{-6b}

^a Observed experimentally, in an operating BWR-3 with MK I containment, data obtained from GE unpublished document, DRF 268-DEV-0009.

^b Assuming 10% of the upper limit values.

^c Release of Cs-137 activity would strongly depend on the core inventory which is a function of fuel burnup.

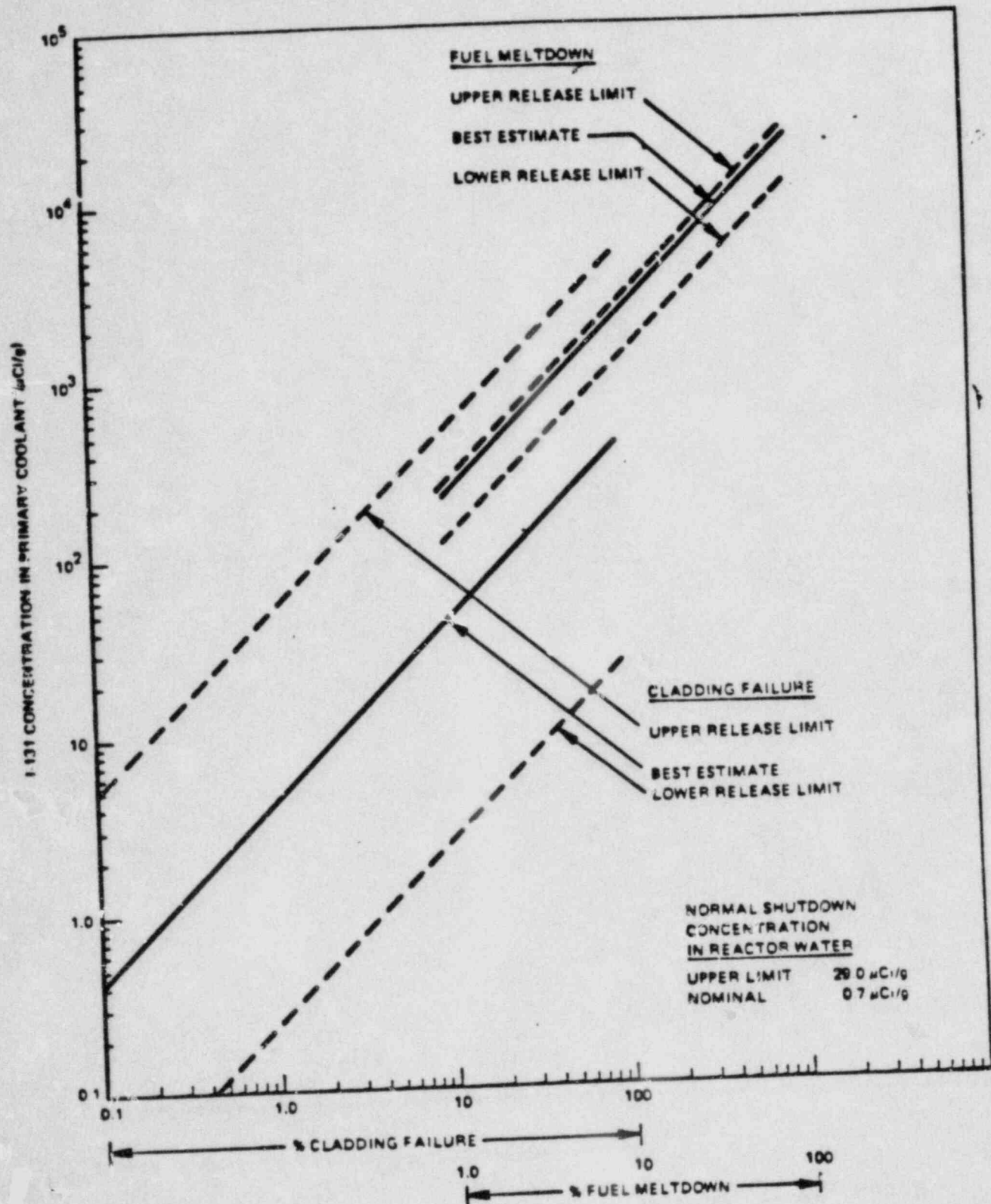


Figure 1. Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

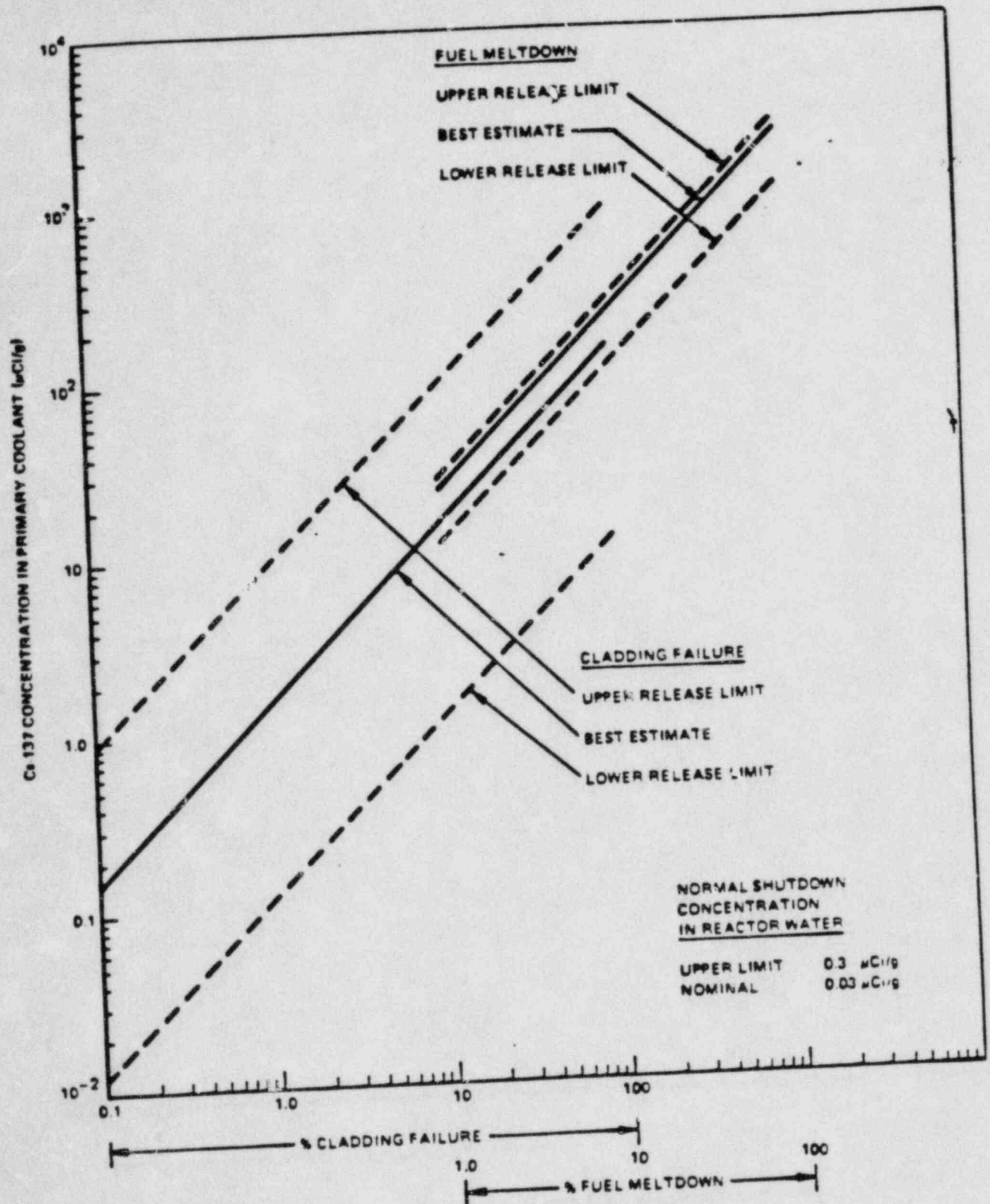


Figure 2. Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

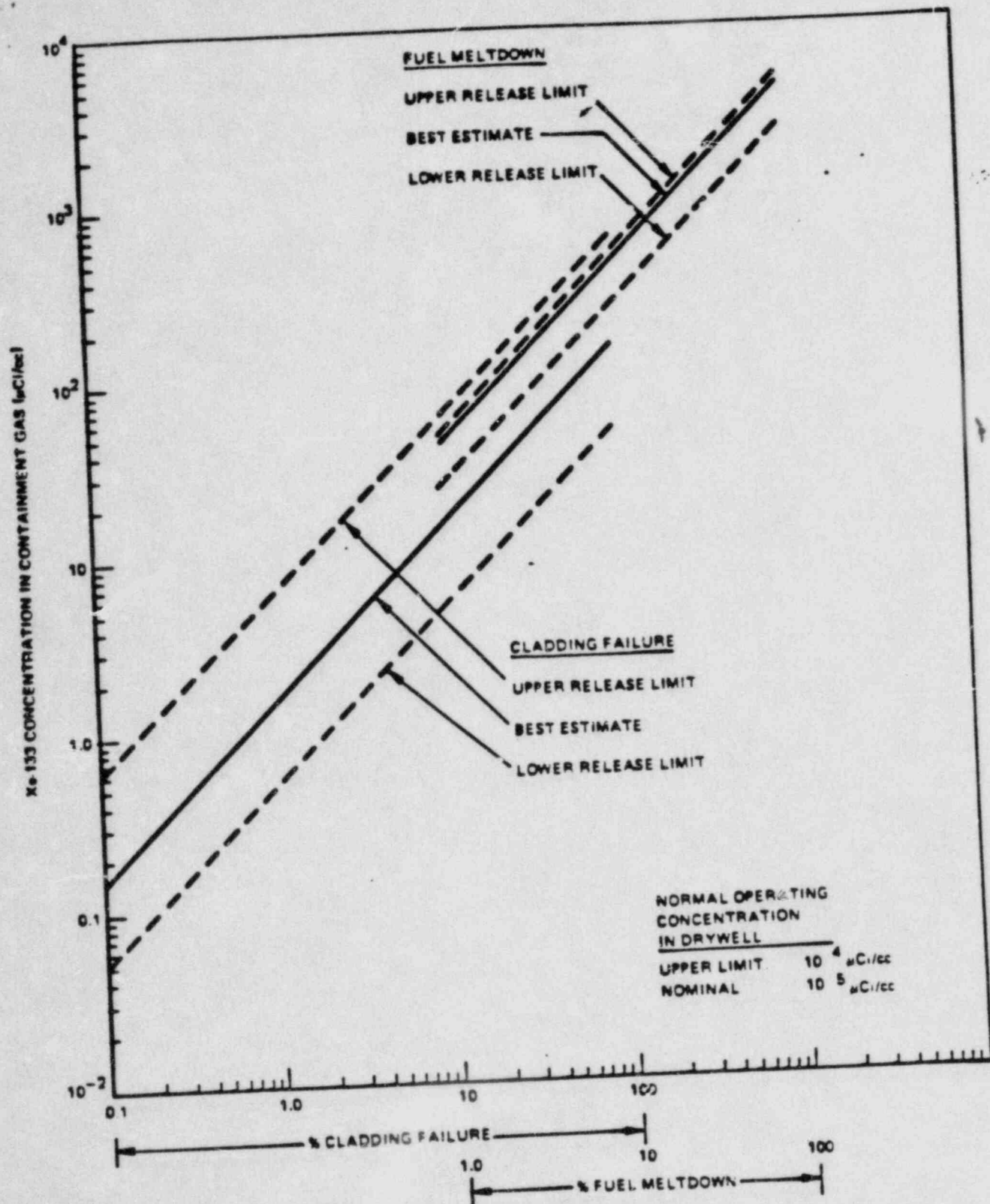


Figure 3. Relationship Between Xe-133 Concentration in the Containment Gas (Drywell + Torus Gas) and the Extent of Core Damage in Reference Plant

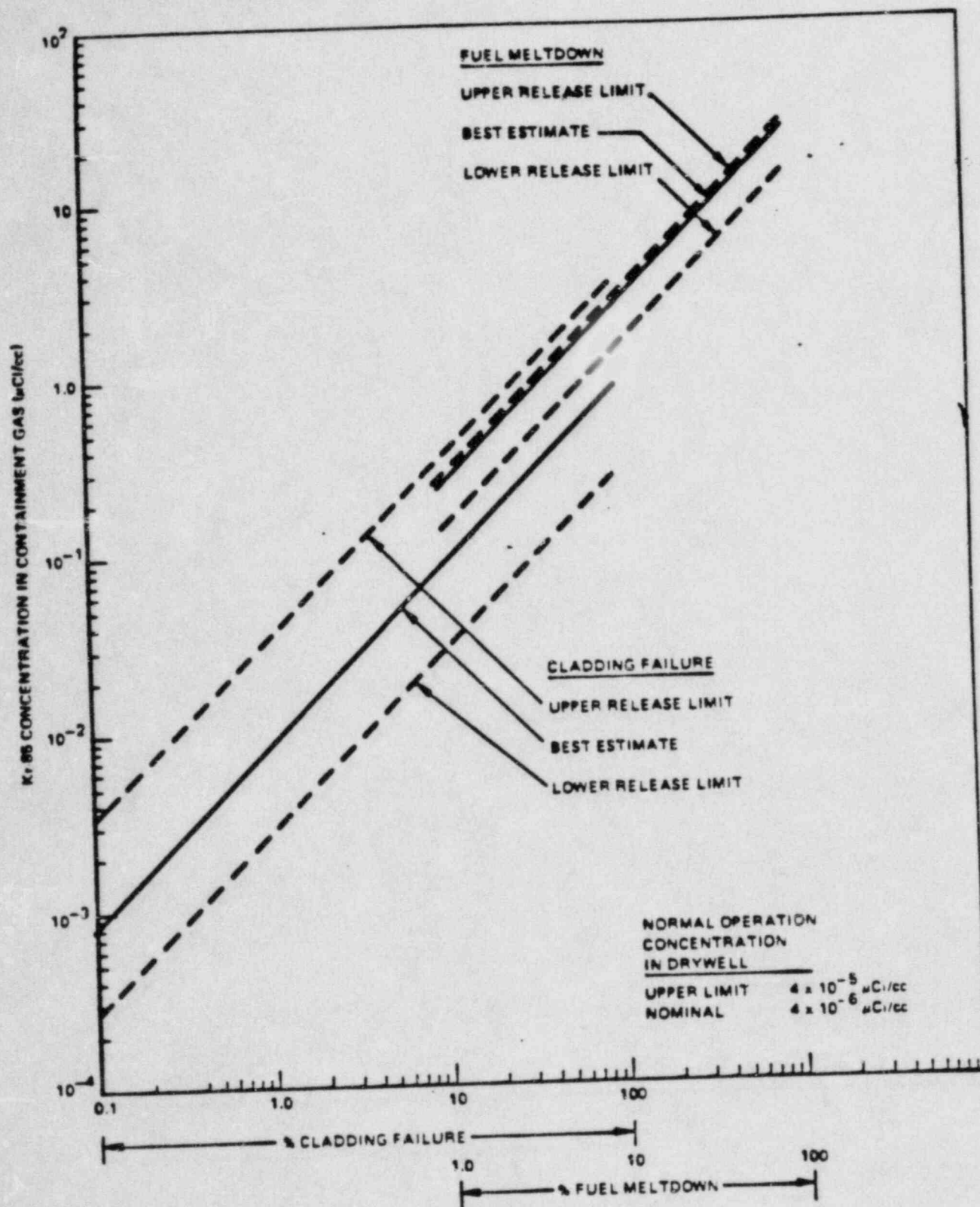


Figure 4. Relationship Between Kr-85 Concentration in the Containment Gas (Drywell + Torus Gas) and the Extent of Core Damage in Reference Plant

If the concentration falls into the range where release of the fission product from the fuel gap or the molten fuel cannot be definitively determined, additional data may be needed to determine the source of fission product release (see below).

It is recommended that both the water and gas phase samples be measured in order to reduce the uncertainty in core damage estimations.

2.1.3 Supplementary Data

In addition to the longer-lived isotopes, some shorter-lived isotope concentrations may be measured in the sample. The ratios of isotopes released from either the fuel gap or the molten fuel are significantly different as shown in Table 3 (see Section 3.3 for details), thus the source (fuel or gap) of release may be identified. Furthermore, some less volatile elements in the core may also start to release as the fuel starts to melt. If the less volatile fission products, such as isotopes of Sr, Ba, La, and Ru (either soluble or insoluble), are found to have unusually high concentrations in the water sample, some degree of fuel melting may be inferred. In a mixture of fission products ^{92}Sr (1.384 MeV) and ^{140}La (1.597 MeV) should be relatively easy to identify and measure from a gamma ray spectrum. More work, however, is needed to establish the baseline concentrations for those isotopes.

2.2 SPECIFIC PLANT APPLICATION

2.2.1 Plant Parameters

The pertinent reactor parameters for selected plants currently being retrofitted with the post accident sampling system are tabulated in Table 4.² Similar information is available for all BWRs.

Table 3
RATIOS OF ISOTOPES IN CORE INVENTORY AND FUEL GAP

<u>Isotope</u>	<u>Half-Life</u>	<u>Activity Ratio* in Core Inventory</u>	<u>Activity Ratio* in Fuel Gap</u>
Kr-87	76.3 m	0.233	0.0234
Kr-88	2.84h	0.33	0.0495
Kr-85m	4.48h	0.122	0.023
Xe-133	5.25d	1.0*	1.0*
I-134	52.6 m	2.3	0.155
I-132	2.3 h	1.46	0.127
I-135	6.61h	1.97	0.364
I-133	20.8 h	2.09	0.685
I-131	8.04d	1.0*	1.0*

*Ratio = $\frac{\text{noble gas isotope concentration}}{\text{Xe-133 concentration}}$ for noble gases

= $\frac{\text{Iodine isotope concentration}}{\text{I-131 concentration}}$ for iodines

Table 4
PLANT PARAMETERS²

Plant	Reactor Type/ Containment Design	Rated Power (MWt)	Primary Coolant*		Containment Gas*	
			Reactor Water Mass (10 ⁸ g)	Suppression Pool Water (10 ⁹ g)	Drywell Gas Volume (10 ⁹ cc)	Torus/ Containment Gas Volume (10 ⁹ cc)
Standard	BWR 6/III	3579	2.46	3.67	7.77	32.5
Brunswick-1/2	BWR 4/I	2436	2.14	2.48	4.65	3.46
Chinshan-1/2	BWR 4/I	1775	1.76	1.93	3.68	2.69
Cofrentes	BWR 6/III	2894	2.04	3.14	6.91	32.43
Cooper	BWR 4/I	2380	2.00	2.48	3.75	3.03
Dresden-2/3	BWR 3/I	2527	2.61	3.18	4.48	3.30
Duane Arnold	BWR 4/I	1593	1.45	1.67	2.67	2.67
Fermi-2	BWR 4/I	3293	2.77	3.23	4.64	3.71
Fitzpatrick	BWR 4/I	2436	2.14	3.00	4.37	3.20
Hanford-2	BWR 5/II	3323	2.74	3.17	5.75	4.08
Hatch-1	BWR 4/I	2436	2.00	2.47	4.07	3.20
Hatch-2	BWR 4/I	2436	2.00	2.47	4.12	3.11
Hope Creek-1/2	BWR 4/I	3293	2.93	3.34	4.79	3.78
Kuo sheng-1/2	BWR 6/III	2894	2.04	3.74	6.74	40.50
Limerick-1/2	BWR 4/II	3293	2.93	3.63	6.66	4.23
Millstone-1	BWR 3/I	2011	2.05	2.78	4.16	3.06
Monticello	BWR 3/I	1670	1.75	1.93	3.80	2.76
NMP-1	BWR 2/I	1850	2.17	2.34	5.10	3.33
Oyster Creek	BWR 2/I	1933	2.05	2.32	5.10	3.85
Peach Bottom-2/3	BWR 4/I	3293	2.67	3.48	4.98	3.62
Pilgrim	BWR 3/I	1998	2.05	2.38	4.16	3.18
Susquehanna-1/2	BWR 4/II	3293	2.92	3.60	6.79	4.36
Vermont Yankee	BWR 4/I	1593	1.77	1.93	3.79	3.18

* Total Primary Coolant Mass = Reactor Water + Suppression Pool Water

Total Containment Gas Volume = Drywell Gas + Torus (or Primary Containment
in Mark III gas)

2.2.2 Procedure

The extent of core damage in an operating BWR can be determined by comparing the measured concentrations of major fission products in either the gas or water samples, after appropriate normalization, with the reference plant data. The following procedure is recommended.

1. Obtain the samples from the post accident sampling system, and the concentration of a fission product i (C_{wi} in water or C_{gi} in gas is determined).
2. Correct the measured concentration for decay to the time of reactor shutdown.
3. Correct the measured gaseous activity concentration for temperature and pressure difference in the sample vial and the containment (torus) gas phase (see footnote on Page 2-12).
4. Calculate the fission product inventory correction factor (Section 2.2.2.2).
5. Calculate the plant parameter correction factor (Section 2.2.2.3).
6. By using the correction factors, calculate the normalized concentration, C_{wi}^{Ref} or C_{gi}^{Ref} (Section 2.2.2.1).
7. Use Figures 1 through 4 to estimate the extent of fuel or cladding damage.

2.2.2.1 Comparison with Reference Plant Data

The extent of core damage can be estimated from the measured fission product concentrations in either the gas or water samples, as described for the reference plant. However, the measured concentration must be corrected

for the differences in operation power level, time of operation, primary coolant mass and containment gas volume.

$$C_{wi}^{Ref} = C_{wi} e^{\lambda_1 t} \times F_{I1} \times F_w$$

or

$$C_{gi}^{Ref} = C_{gi} e^{\lambda_1 t} \times F_{I1} \times F_g$$

where

C_{wi}^{Ref} = concentration of isotope i in the reference plant coolant ($\mu\text{Ci/g}$)

C_{gi}^{Ref} = concentration of isotope i in the reference plant containment gas ($\mu\text{Ci/cc}$)

C_{wi} = measured concentration of isotope i in the operating coolant at time, t ($\mu\text{Ci/g}$)

C_{gi} = measured concentration of isotope i in the operating containment gas at time, t ($\mu\text{Ci/cc}$)*

$e^{\lambda_1 t}$ = decay correction to the time of reactor shutdown

λ_1 = decay constant of isotope i (day^{-1})

*The following correction for the measured concentration is needed if the temperature and pressure in the sample vial (T_1, P_1) are different from that in the containment (T_2, P_2):

$$C_{gi} = C_{gi(\text{vial})} \times \frac{P_2 T_1}{P_1 T_2}$$

t = time between the reactor shutdown and the sample time (day)

F_{Ii} = inventory correction factor for isotope i (see Section 2.2.2.2)

F_g = containment gas volume correction factor (see Section 2.2.2.3)

F_w = primary coolant mass correction factor (see Section 2.2.2.3)

2.2.2.2 Inventory Correction Factor

$$F_{Ii} = \frac{\text{Inventory in reference plant}}{\text{Inventory in operating plant}}$$

$$= \frac{3651 (1 - e^{-1095 \lambda_i})}{\sum_j \left[P_j (1 - e^{-\lambda_i T_j}) e^{-\lambda_i T_j^0} \right]}$$

where

P_j = steady reactor power operated in period j (MWt)*

T_j = duration of operating period j (day)*

T_j^0 = time between the end of operating period j and time of the last reactor shutdown (day)

For a particular short-lived isotope, i , a calculation for only a period of ~ 6 half-lives of reactor operation time before reactor shutdown should be accurate enough. An example of calculation is illustrated in Appendix A. It should be pointed out that the computer calculation of core inventory takes into account the fuel burnup, plutonium fission and neutron capture reactors.

*In each period, the variation of steady power should be limited to $\pm 20\%$.

The correction factor calculated from this equation may not be entirely accurate, but the error is insignificant in comparison to the uncertainties in the fission product release fractions (Table 5) and other assumptions (Section 3.2).

2.2.2.3 Plant Parameter Correction Factors

$$F_w = \frac{\text{operating plant coolant mass (g)}}{\text{reference plant coolant mass (3.92 x 10}^9 \text{ g)}} = \frac{2.54 \times 10^9}{3.92 \times 10^9} = .648$$

$$F_g = \frac{\text{operating plant containment gas volume (cc)}}{\text{reference plant containment gas volume (4 x 10}^{10} \text{ cc)}} = \frac{.734 \times 10^{10}}{4 \times 10^{10}} = .1835$$

In case the fission product concentrations are measured separately for the reactor water and suppression pool water or the drywell gas and the torus gas, the measured concentrations C_{wi} or C_{gi} would be averaged from the separate measurements:

$$C_{wi} = \frac{(\text{conc. in Rx water}) \times (\text{Rx water mass}) + (\text{conc. in pool}) \times (\text{pool water mass})}{\text{Reactor water mass} + \text{pool water}}$$

$$C_{gi} = \frac{(\text{conc. in drywell}) \times (\text{drywell gas vol}) + (\text{conc. in torus}) \times (\text{torus gas vol})}{\text{drywell gas volume} + \text{torus gas volume}}$$

Table 5
BEST-ESTIMATE FISSION PRODUCT RELEASE FRACTIONS^{7,8}

	Gap Release			Meltdown Release			Oxidation Release			Vaporization Release		
	Nominal	Lower Limit	Upper Limit	Nominal	Lower Limit	Upper Limit	Nominal	Lower Limit	Upper Limit	Nominal	Lower Limit	Upper Limit
Noble Gases (Xe, Kr)	0.030	0.010	0.12	0.873	0.485	0.970	0.087	0.078	0.097	0.010	0.010	0.010
Halogens (I, Br)	0.017	0.001	0.20	0.885	0.492	0.983	0.088	0.078	0.098	0.010	0.010	0.010
Alkali Metals (Cs, Rb)	0.050	0.004	0.30	0.760	0.180	0.855	---	---	---	0.190	0.190	0.190
Tellurium Group (Te, Se, Sb)	0.0001	3×10^{-7}	0.04	0.150	0.05	0.250	0.510	0.340	0.680	0.340	0.340	0.340
Noble Metals (Ru, Rh, Pd, Mo, Tc)	---	---	---	0.030	0.01	0.10	0.873	0.776	0.970	0.005	0.001	0.024
Alkaline Earths (Sr, Ba)	1×10^{-6}	3×10^{-9}	0.0004	0.100	0.02	0.20	---	---	---	0.009	0.002	0.045
Rare Earths (Y, La, Ce, Nd, Pr, Eu, Pm, Sm, Np, Pu)	---	---	---	0.003	0.001	0.01	---	---	---	0.010	0.002	0.050
Refractories (Zr, Nb)	---	---	---	0.003	0.001	0.01	---	---	---	---	---	---

3. TECHNICAL BASIS

3.1 FISSION PRODUCT CONCENTRATIONS IN THE PRIMARY SYSTEM DURING REACTOR SHUTDOWN UNDER NORMAL OPERATION CONDITIONS

3.1.1 Fission Product Concentrations in Reactor Water

It is well known that some volatile and water soluble fission products, mainly iodine and cesium isotopes, will be released (called spiking) from defect fuel rods when the reactor is shutdown and depressurized. Based on Pasedag of NRC,³ the maximum I-131 release would be 10 Ci per each $\mu\text{Ci/sec}$ release rate during normal power operation (see Appendix B). According to the GE design basis of I-131 release rate at 700 $\mu\text{Ci/sec}$ ⁴ a maximum of 7000 Ci of I-131 may be released during reactor shutdown, and the concentration in reactor water would be 29 $\mu\text{Ci/g}$.

An analytical model to predict the magnitude of I-131 spiking following reactor shutdown in operating BWRs has been reported by Brutschy et al.⁵ The "best estimate" concentration for I-131 has to be calculated based on the analytical mode⁵ for the individual reactor according to its fuel condition (see Appendix C). However, if one adopts a standard I-131 concentration of 5×10^{-3} $\mu\text{Ci/g}$ or ~ 18 $\mu\text{Ci/sec}$ as proposed by ANS⁶, the nominal I-131 spiking is estimated to be ~ 0.7 $\mu\text{Ci/g}$ in the reference plant water. This concentration is consistent with an average spiking concentration observed experimentally (see Appendix C, Figure C-1).⁵ The results of these estimations, including the Cs-137 concentration, have been summarized in Table 2.

Potential future research in this area will be discussed in Section 4.

3.1.2 Noble Gas Concentrations in Drywell and Torus Gas Phase

Similar to the spiking magnitude, the noble gas activities in the drywell and the torus gas may vary significantly from reactor to reactor, mainly depending on the fuel condition and the steam leakage rate. In an operating BWR when the Xe-133 release rate measured at the steam jet air ejector (SJAE)

was 1.5×10^4 $\mu\text{Ci/sec}^*$ (compared to design basis release rate at 8200 $\mu\text{Ci/sec}$). the noble gas concentrations in the drywell were determined to be $\sim 10^{-4}$ $\mu\text{Ci/cc}$ for Xe-133 and $\sim 4 \times 10^{-5}$ $\mu\text{Ci/cc}$ for Kr-85. These data may be considered as the upper limit values.

3.2 FISSION PRODUCT RELEASE SOURCE TERMS UNDER ACCIDENT CONDITIONS

The source terms for the damaged core under accident conditions have been proposed by several investigators.^{7,8} The "best estimate" release source terms for different chemical groups of fission products are summarized in Table 5.

The release of fission products from the damage core has been estimated to be a function of temperature,⁹ and time after the loss-of-coolant accident.⁷ In the present procedure, the fraction of fission product release from the core is assumed to be proportional to the fraction of core damage as suggested by Malinauskas, et. al.⁹ It is further assumed that the core is homogeneous so that each fuel rod has an identical exposure history. The fuel cladding rupture has been assumed to occur over the temperature range from about 780° to 1100°C,⁹ and the entire fission product noble gas inventory in the fuel gap would be released. All other fission products in the fuel gap, which may be present in a condensed phase, or as vapor in equilibrium with a condensed phase, will not be released as quickly as noble gases until the temperature is further increased. According to a model calculation,⁷ portions of the fuel may start to melt before the cladding is totally destroyed.

3.3 ISOTOPIC DISTRIBUTION IN FUEL GAP

Diffusion equations predict that the fractional release of radioactive isotopes from the fuel to the plenum and void spaces should be inversely proportional to the square root of the decay constant for isotope reaching production-decay equilibrium.^{10,11} This prediction has been substantiated by experimental data reported by several investigators.¹²⁻¹⁷ A comparison of

*Data obtained from GE unpublished document, DRF 268-DEV00009. The fission product release pattern was found to be mostly "recoil."

isotopic distributions in the total fuel inventory and the predicted distribution for some major fission products has been shown in Table 3. Thus, by measuring the ratios of fission product activities in either the gas or water samples, the source of fission product release may be semi-quantitatively determined (see more discussion in Section 4).

3.4 ANTICIPATED CHEMICAL BEHAVIOR OF IODINE AND COOLANT CHEMISTRY UNDER ACCIDENT CONDITIONS

The results of measurements of Three Mile Island-2 (TMI-2)^{18,19} indicate that the airborne radioiodine release was much lower when compared to the noble gas activity release (by a factor of $\sim 10^6$). Extensive investigations at the Oak Ridge National Laboratory (ORNL) on the nature and quantity of fission product release from the over-heated fuel have concluded that cesium iodide CsI (B.P. = 1280°C) is the primary volatile species released from the fuel at elevated temperatures.²⁰ The behavior of iodine under loss-of-coolant accident (LOCA) conditions has been evaluated by Lin²¹ and Campbell et. al.²²

For iodine at a concentration of a few ppm in aqueous solutions, the redox reactions should be more predictable and formations of anomalous or organic species should be much smaller than that at very low concentrations as generally assumed for radioiodine release. If iodine is released as CsI, it would stay in water as the I^- ion in a slightly basic solution (mainly due to Cs ions which may be released as elemental Cs or Cs oxides in addition to CsI). Air oxidation²³ or radiation-induced oxidation of I^- to I_2 ²⁴ is not very likely to occur in a basic solution. In addition to the reducing nature of zirconium and iron metals in the core, the production of hydrogen from Zr steam reactions should make the chemistry environment in the primary system favorable to reducing reactions for iodine.

There are at least three known volatile forms of iodine, I_2 , HIO, and organic iodine. The formation of I_2 from I^- is not very likely in basic solutions. The existence of HIO has never been chemically identified due to its low stable concentration. The airborne species called HIO is one which behaves differently from I_2 and organic iodine determined by using the iodine species

sampling method developed by Keller, et. al.²⁵ However, some convincing evidence has been given by Lin²⁶ that HIO, a product of I_2 hydrolysis, is the second volatile inorganic species in the gas phase when I_2 was initially added to water in equilibrium partitioning studies. The partition coefficient increases with decreasing iodine concentration; at very low iodine concentrations, the total iodine partition coefficients have been determined to be ~ 8000 at 21°C and ~ 1600 at 72°C .²⁶ It must be pointed out that since both I_2 and HIO are very reactive species, any reducing impurities in water or on construction material surfaces would reduce I_2 or HIO to I^- and significantly reduce the airborne iodine concentration.

The mechanisms of converting inorganic iodine to organic iodine, which is generally observed in gas phase at very low concentrations, are largely unknown. It is certain, however, that at least more than a stoichiometric amount of organic species (or carbon-containing compounds) should be readily available for reaction with iodine. As such organic species are limited, the results of several experiments²⁷ indicate that the yield of organic iodine decreases with increasing iodine concentration in the gas phase. Less than 0.1% conversion is expected when the airborne iodine concentration is 1 g/m^3 or larger.²⁷ The total iodine concentration could be $\sim 3 \text{ g/m}^3$ in the containment free air space if all iodine is assumed to become airborne. It is also important to realize that the organic iodine, e.g., CH_3I is readily hydrolyzed in water²⁸ and basic solutions²⁹ at higher temperatures. The half-time of hydrolysis is ~ 20 min in water at 100°C and ~ 3 sec at 200°C , based on Heppollette and Robertson's data.²⁸

It is obvious that the very low release of iodine activities to the atmosphere in the TMI-2 accident can be easily explained in terms of the nature of iodine released from the fuel and the subsequent stabilization in water. Water plays an important role in preventing iodine from release to the atmosphere. In the present procedure, all the iodine activities are assumed to stay in water, and the airborne activities are dominated by the noble gas fission products.

The chemistry in the primary coolant may be significantly changed under accident conditions. Mainly due to the release of cesium, the water pH may increase to $\sim 10.5^{21}$ and the water conductivity may increase from ~ 10 $\mu\text{S}/\text{cm}$ (torus water quality specification) to as high as ~ 170 $\mu\text{S}/\text{cm}$.

4. DISCUSSION AND SUGGESTIONS FOR FUTURE WORK

It is obvious that the uncertainty of gas release fractions for iodine and cesium are too large for an accurate calculation of the extent of core damage. While additional experimental work in fuel gap measurements is apparently needed, the lower limit release fraction for iodine may be reevaluated by examining the iodine spiking release data from defective fuel rods following normal operation shutdowns.

Although the I-131 spiking data has been well documented previously⁵, the analytical model may be refined to reflect more recent experimental data. The maximum spiking release of I-131 estimated by Pasedag³, based on pre-1973 data, is obviously too high, particularly when the improved fuels which are currently used in most of the operating BWRs are considered.

The accuracy of core damage estimation may be significantly improved by measuring more than iodine, cesium, and noble gas activities. Some less volatile but easy to measure isotopes of Sr, Ba, La, and Ru may be determined in the water sample. More work, however, is needed to establish the release fractions as well as the baseline (shutdown spiking) concentrations for those isotopes. Additionally, some chemical analysis data, such as hydrogen production from water-zirconium reaction may be used to improve the estimate. The degree of core damage between fuel cladding failures and core melt, namely fuel overheating, may be determined.

As mentioned in Section 3.3, it is possible to determine the source of fission product release by measuring the activity ratios of noble gases or iodine isotopes. It must be cautioned, however, each isotope should be accurately measured. Particular care must be exercised when the Xe-133 activity is determined in a mixture of other fission products with high concentration because of its low gamma ray energy (81 keV). Additional work is required to perfect this procedure.

5. REFERENCES

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Appendix A

SAMPLE CALCULATION OF FISSION PRODUCT INVENTORY CORRECTION FACTOR

$$F_{Ii} = \frac{\text{Inventory of nuclide } i \text{ in reference plant}}{\text{Inventory of nuclide } i \text{ in operating plant}}$$

$$= \frac{3651 (1 - e^{-1095 \lambda_i})}{\sum_j \left[P_j (1 - e^{-\lambda_i T_j}) e^{-\lambda_i T_j^0} \right]}$$

where

P_j = steady reactor power operated in period j (MWt)

λ_i = decay constant of nuclide i (day^{-1})

T_j = duration of operating period j (day)

T_j^0 = time between the end of operating period j and time of last reactor shutdown (day)

3651 = ave. operation power (in MWt) for the reference plant.

1095 = continuous operation time (in day) for the reference plant.

Assuming a reactor has the following power operation history:

Operation Period	Days Since Startup	Operation Time T_j (day)	T_j^0	Average Power P_j (MWt)
1A	1 - 60	60	254	1000
1B	61 - 70	---	---	0
2A	71 - 270	270	44	2000
2B	271 - 300	---	---	0
3	301 - 314	14	0	3000

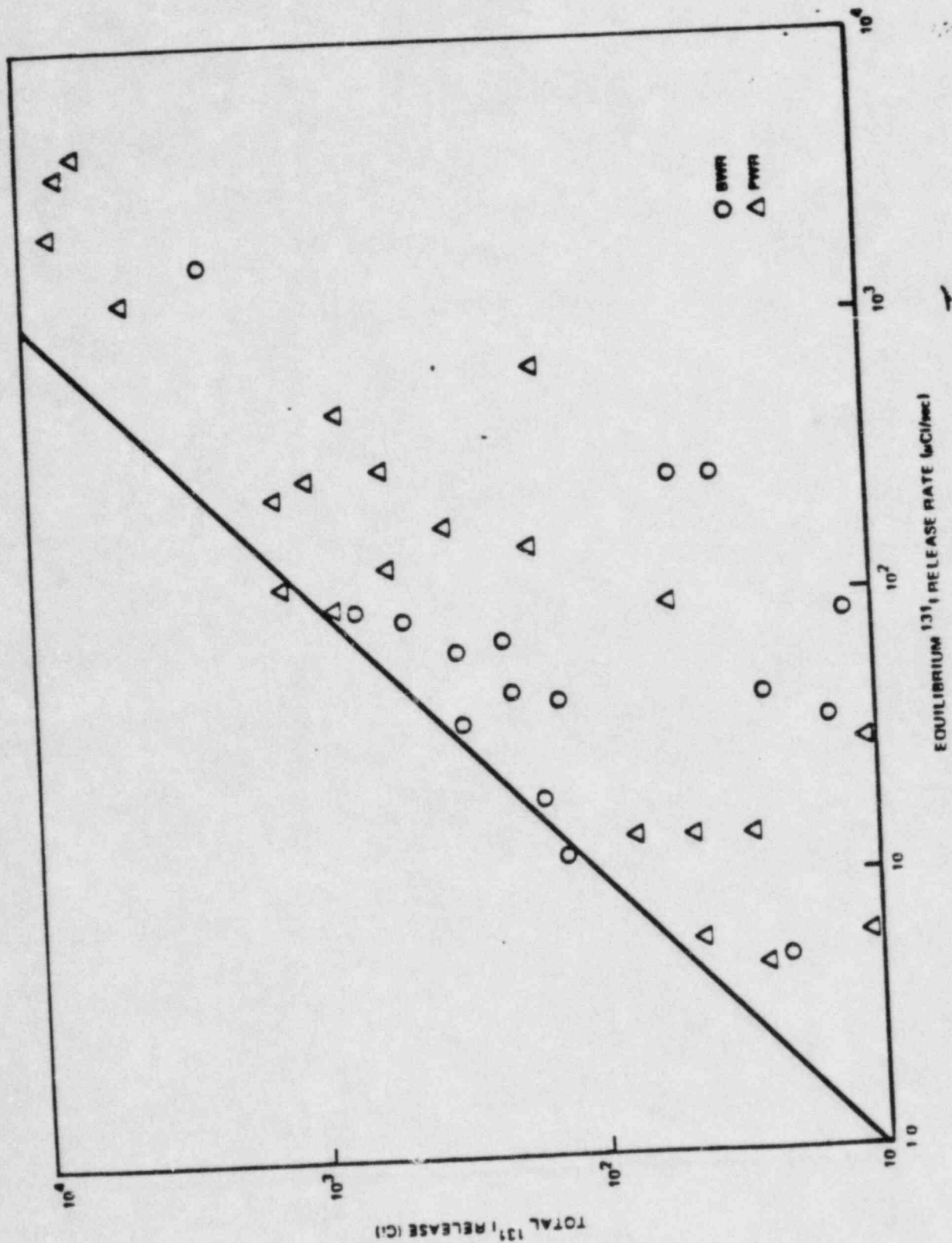
- For I-131 ($\lambda = 0.0862 \text{ day}^{-1}$)

$$\begin{aligned}
 F_{I(I-131)} &= \frac{3651(1-e^{-0.0862 \times 1095})}{1000(1-e^{-0.0862 \times 60})e^{-0.0862 \times 254} + 2000(1-e^{-0.0862 \times 200})e^{-0.0862 \times 44} + 3000(1-e^{-0.0862 \times 14})e^{-0.0862 \times 0}} \\
 &= \frac{3651}{20 + .45 + 2103} = 1.7
 \end{aligned}$$

- For Cs-137 ($\lambda = 6.29 \times 10^{-5} \text{ day}^{-1}$)

$$\begin{aligned}
 F_{I(Cs-137)} &= \frac{3651(1-e^{-6.29 \times 10^{-5} \times 1095})}{1000(1-e^{-6.29 \times 10^{-5} \times 60})e^{-6.29 \times 10^{-5} \times 254} + 2000(1-e^{-6.29 \times 10^{-5} \times 200})e^{-6.29 \times 10^{-5} \times 44} + 3000(1-e^{-6.29 \times 10^{-5} \times 14})e^{-6.29 \times 10^{-5} \times 0}} \\
 &= \frac{243.16}{3.74 + 24.93 + 2.64} = 7.77
 \end{aligned}$$

Appendix B
TOTAL I-131 RELEASE DURING A SPIKING SEQUENCE⁴



Appendix C

ESTIMATION OF MAXIMUM IODINE-131 SPIKE CONCENTRATION IN REACTOR WATER

The magnitude of iodine spiking in BWRs can be predicted by an empirical relationship proposed by Brutschy, et. al.⁵ The data basis for the empirical relationship are shown in Figure C-1.

In order to predict the maximum I-131 concentration in reactor water (assuming no reactor water clean-up system in operation) during a shutdown spiking, the following information should be known for the reactor during steady state operation: I-131 concentration in reactor water, I-131 release source term, and the knowledge of the fission gas release characteristics.

The fission product release from the defective fuel rods in a BWR during steady operation is empirically characterized by

$$A_1 (\text{Ci/sec}) = KY_1 \lambda_1^{1-b}$$

or

$$R_1 (\text{fission/sec}) = \frac{A_1}{Y_1 \lambda_1} = K \lambda_1^{-b}$$

where

R_1 = fission product release source term

K = a dimensional constant establishing the level of release

b = a dimensionless constant establishing the relative amount of each nuclide in the mixture of similar chemical groups, i.e., noble gases or iodine isotopes.

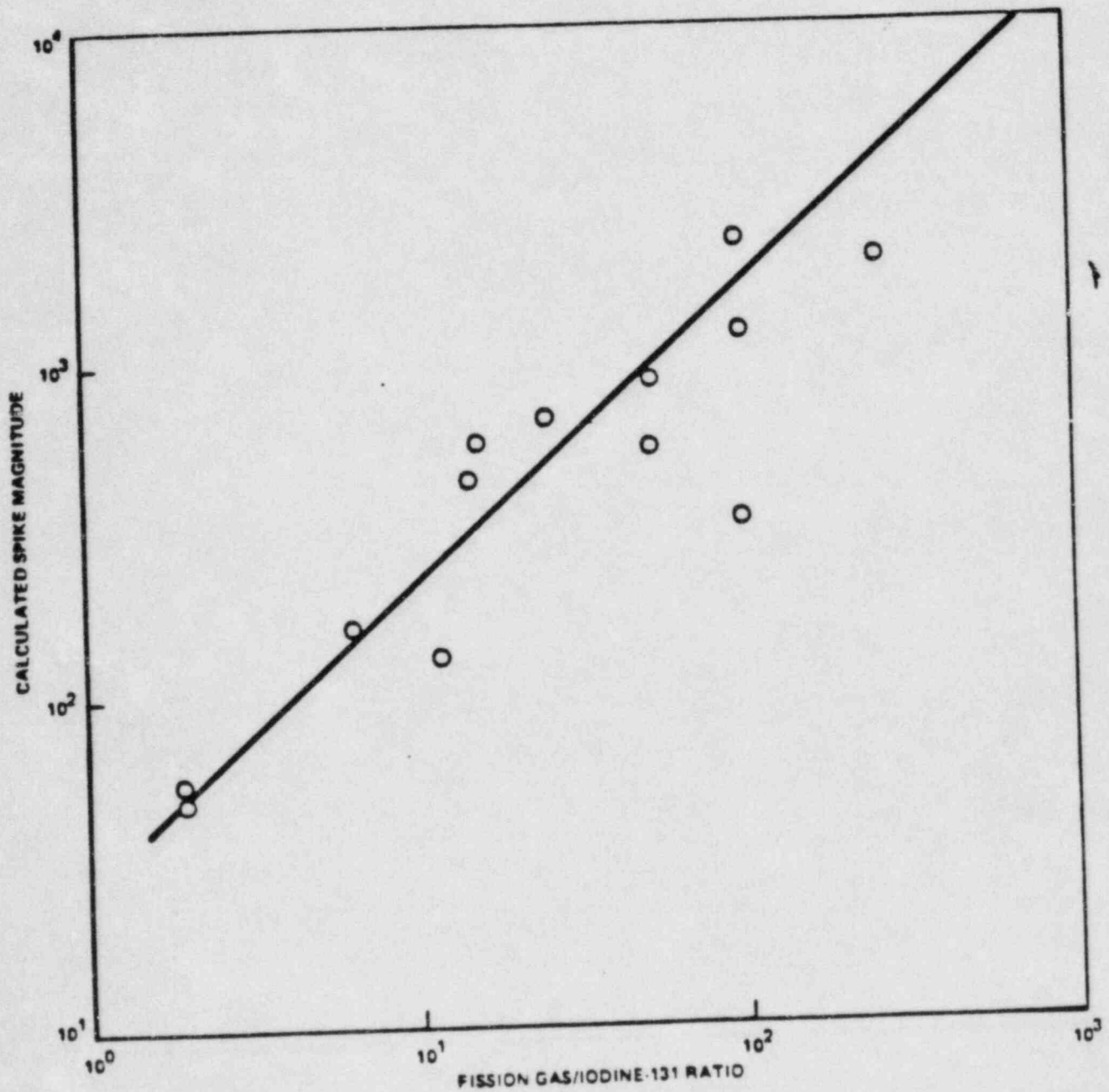


Figure C-1. Fission Gas to Iodine-131 Ratio versus Calculated Spike Magnitude⁵
(Assuming one peak, and cleanup system out of service)

By plotting $A_1/Y_1\lambda_1$ against λ_1 for noble gases or iodine isotopes on a log-log paper, a straight line can be obtained with a negative slope of b as demonstrated in Figure C-2.

Experimentally, the noble gas activity release rate (concentration in the offgas times offgas flow rate) is measured at the steam jet air ejector (SJAE) sample point. The measurement of iodine activity release rate is more complicated. By assuming no iodine activity is returned from the feedwater into the reactor vessel.* The release rate can be calculated by

$$A_1 = C_1 W (\lambda_1 + \beta_c + \beta_s)$$

where

A_1 = release rate to coolant from the core, $\mu\text{Ci/sec}$.

W = reactor water mass, Kg

λ_1 = decay constant of species 1, sec^{-1}

β_c = reactor water cleanup (RWCU) system removal time constant, sec^{-1} , which is defined as $\beta_c = f/W$, assuming ~100% efficient.

f = RWCU flow rate, Kg/sec

β_s = steam removal time constant, sec^{-1} , which is defined as $\beta_s = cF/W$

c = iodine carryover, defined as the ratio of

$$\frac{\text{the concentration of species 1 in condensate}}{\text{the concentration of species 1 in reactor water}}$$

F = steam flow rate, Kg/sec

* This is true only for non-forward pumping plants.

The source term ratio (R_g/R_{I-131}) in Figure C-1 is defined as the ratio of the noble gas source term estimated at the λ of I-131 to the I-131 source term (see Figure C-2). By knowing R_g/R_{I-131} and the magnitude of I-131 spiking (maximum spiking concentration/steady state concentration) from Figure C-1, the maximum I-131 concentration in reactor water during a shutdown spiking can be estimated.

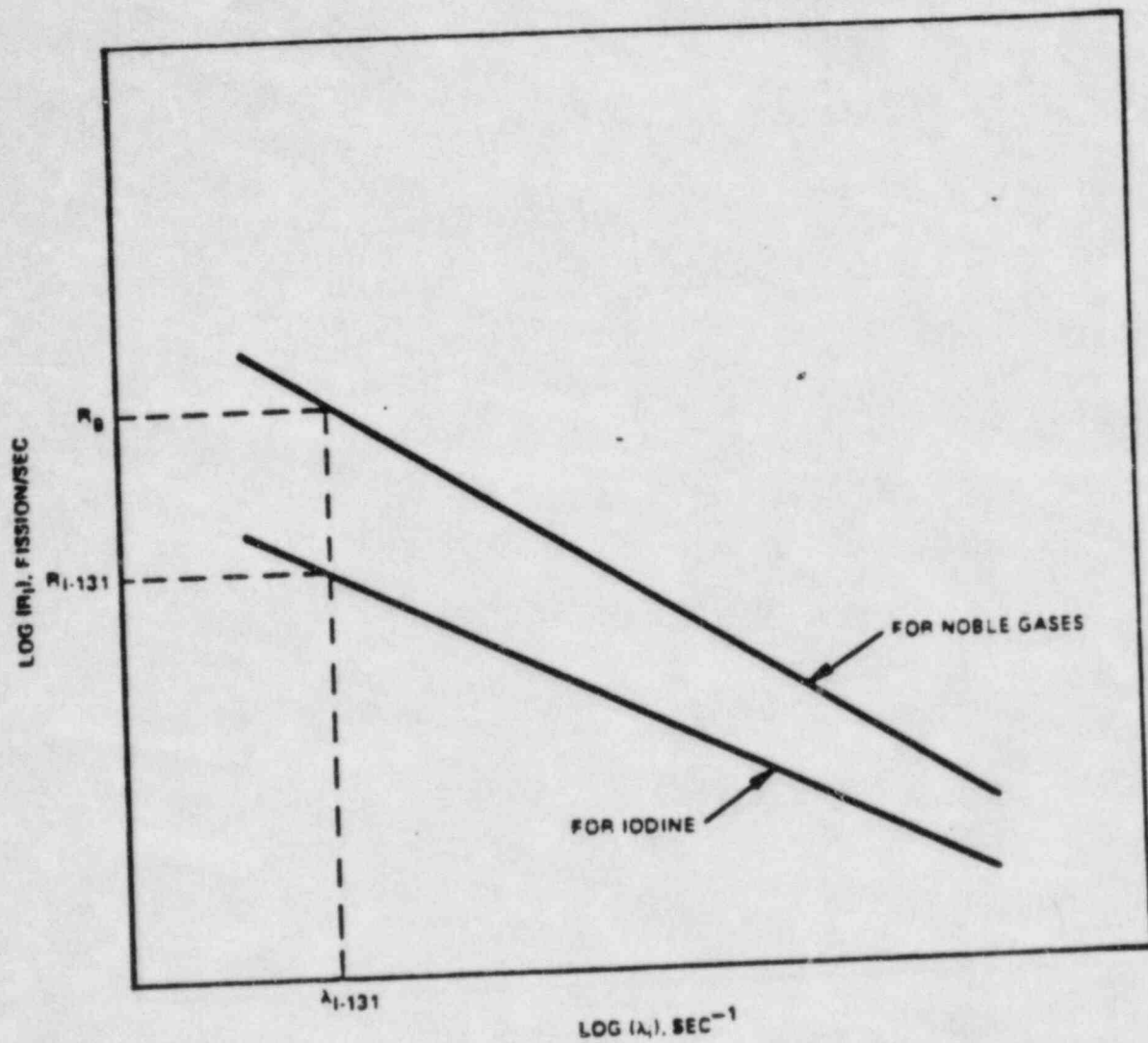


Figure C-2. Plot of $\text{Log } (R_i)$ versus $\text{Log } (\lambda_i)$

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ATTACHMENT 2

INTEGRATION OF OTHER PLANT
PARAMETERS INTO CORE DAMAGE
ESTIMATE

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1.0 INTEGRATION OF "OTHER PLANT PARAMETERS" INTO CORE DAMAGE ESTIMATE

1.1 Purpose and Scope

The purpose of this section is to address NRC Clarification 2(a)2 regarding the integration of other plant parameters into the determination of an estimate of core damage for a suspected degraded core event. This additional information would provide verification of the initial estimate of core damage based on radionuclide measurements using the post-accident sampling system (PASS).

The procedure for estimation of core damage based on radionuclide measurements from the PASS has been previously described and provided to those utilities which use that system¹. That procedure involves calculations of fission product inventories in the core and the release of those fission products into the primary system under postulated loss-of-coolant (LOCA) conditions. For that procedure, a BWR-6/238 with a Mark III containment is used as the reference plant. For that plant, plots of core damage estimate versus radionuclide concentrations in reactor water and containment atmosphere are provided. The procedure describes the method for determining the extent of core damage for each unique plant by comparing the measured concentrations of major fission products in either gas or water samples, after appropriate normalization, with the reference plant data. Consequently, core damage estimates can be made for each unique plant based on radionuclide measurements using the PASS.

1.2 Identification of Other Significant Parameters

There are several other plant parameters which are measured in the BWR which can provide sufficient information to confirm his initial core damage estimate based on radionuclide measurements.

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For some of these parameters, correlations similar to that which is provided for the radionuclide measurements can be developed which provide confirmation of the initial core damage estimate. Such correlations can be developed for the parameters of containment radiation level and containment hydrogen level.

Containment radiation level provides a measure of core damage, because it is an indication of the inventory of airborne fission products (i.e., noble gases, a fraction of the halogens and a much smaller fraction of the particulates) released from the fuel to the containment. Containment hydrogen levels, which are measurable by the PASS or the containment gas analyzers, provide a measure of the extent of metal water reaction which, in turn, can be used to estimate the degree of clad damage.

Another significant parameter for the estimation of core damage is reactor vessel water level. This parameter is used to establish if there has been an interruption of adequate core cooling. Significant periods of core uncover, as evidenced by reactor vessel water level readings, would be an indicator of a situation where core damage is likely. Water level measurement would be particularly useful in distinguishing between bulk core damage situations caused by loss of adequate cooling to the entire core and localized core damage situations caused by a flow blockage in some portion of the core.

There are other parameters which may provide an indication that a core damage event has occurred. These are main steam line radiation level and reactor vessel pressure. The usefulness of main steam line radiation measurement is limited because the main steam line radiation monitors are downstream of the main steam isolation valves (MSIVs) and would be unavailable following vessel isolation. Reactor vessel pressure measurement would provide an ambiguous indication of core damage, because, although a high reactor vessel pressure may be indicative of a core damage event, there are many non-degraded core events which could also result in high reactor vessel pressure.

There are other measurements besides radionuclide measurements which are obtainable using the PASS which would further aid in estimating core damage. As noted in the procedure already supplied, detection of such elements in the reactor coolant as Sr, Ba, La and Ru is evidence of fuel melting. These indications could be factored into the final core damage estimate.

1.3 Application of Other Significant Parameters to Core Damage Estimate

As noted in Section 1.1, procedures have already been developed which provide an estimate of core damage based on radionuclide measurements. Based on these procedures, an initial assessment of core damage is made. Based on a clarification provided by the NRC, that assessment would appear in a matrix as follows:

Degree of Degradation	Minor (<10%)	Intermediate (10%-50%)	Major (>50%)
No fuel damage	← 1 →		
Cladding Failure	2	3	4
Fuel Overheat	5	6	7
Fuel Melt	8	9	10

As recommended by the NRC, there are four general classes of damage and three degrees of damage within each of the classes except for the "no fuel damage" class. Consequently, there are a total of 10 possible damage assessment categories. For example, Category 3 would be descriptive of the condition where between 10 and 50 percent of the fuel cladding has failed. Note that the conditions of more than one category could exist simultaneously. The objective of the final core damage assessment procedure is to narrow down, to the maximum extent possible, those categories which apply to the actual in-plant situation.

The initial core damage assessment based on radionuclide measurement will provide one or several candidate categories which most likely represent the actual in-plant condition. The other parameters should then be evaluated, as identified in Section 1.2, to corroborate and further refine the initial estimate.

For example, fission product measurement using PASS may indicate Category 4 core damage and, additionally, the potential for fuel overheating and fuel melt (i.e., Categories 5 through 10). Measurement of hydrogen in containment and use of the hydrogen correlation provided in Appendix A of this report could be used to verify that extensive clad damage had occurred. Use of the containment radiation monitor reading along with the correlation provided in Appendix B of this report would verify that a significant fission product release to the containment had occurred, further verifying the initial assessment.

Further analysis of the PASS samples for concentrations of Ba, Sr, La and Ru and consideration of the relative amounts of fission products released would indicate if any fuel melt had occurred.

The flow diagram in Figure 1 indicates how the analysis of the other significant parameters relates to the estimation of core damage based on radionuclide measurements. As noted earlier, Appendices A and B provide correlations for the determination of the degree of core damage based on containment hydrogen and radiation levels.

1.4 Determination of Sample Location

In order to assure a representative sample which reflects the actual in-core condition, care must be taken in selecting a suitable sample location. The selection of a sample location should account for the type of event which will determine where the fission products will concentrate.

For gas sampling, the recommended sampling locations are as follows:

<u>Event Type</u>	<u>Sample Location</u>
Non-Breaks (e.g., MSIV Closure)	Suppression Pool Atmosphere
Small Breaks	Drywell (before depress.) Suppression Pool Atmosphere (after depress.)
Large Breaks (liquid or steam) in Containment	Drywell
Large Breaks outside containment	Suppression Pool Atmosphere

For liquid sampling, the optimum sample point for all events is the jet pumps as long as there is sufficient reactor pressure to provide a sample from that location. If there is not sufficient reactor pressure to allow a sample to be taken from the jet pumps, then the sample should be taken from the sample point in the RHR system.

In order to ensure a representative liquid sample from the jet pumps at low (<1%) power conditions for small break or non-break events, the reactor water level should be raised to the level of the moisture separators. This will fully flood the moisture separators and will provide a thermally induced recirculation flow path for mixing. Several requisition plant licensees have already committed to the NRC to perform this procedure.

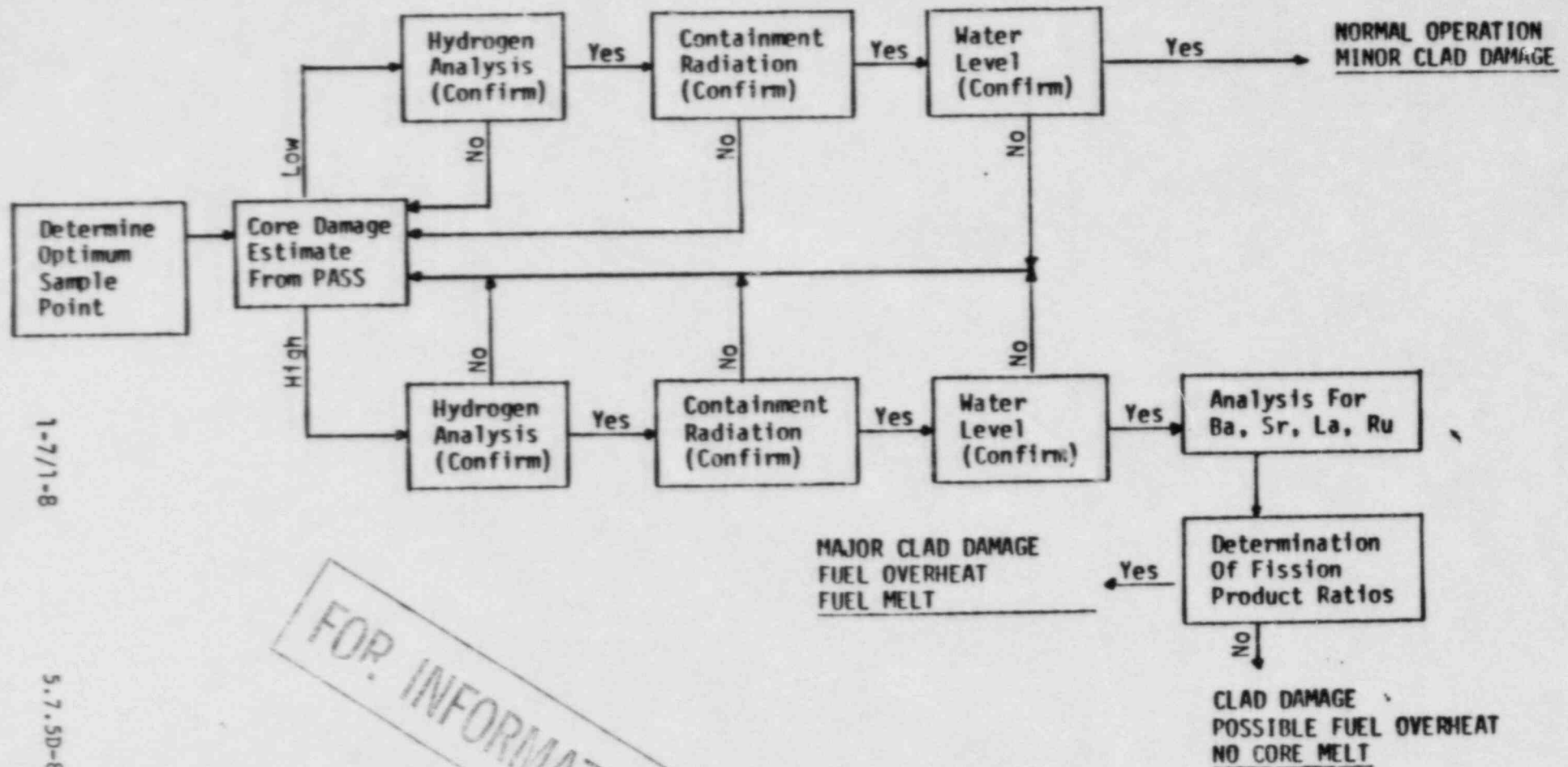
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1.5 References

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FIGURE 1-1
SEQUENCE OF ANALYSIS FOR
ESTIMATION OF CORE DAMAGE



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APPENDIX A

INTEGRATION OF CONTAINMENT ATMOSPHERE
HYDROGEN MEASUREMENT INTO
CORE DAMAGE ESTIMATE

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SUMMARY

The extent of fuel clad damage as evidenced by the extent of metal-water reaction can be estimated by determination of the hydrogen concentration in the containment. That concentration is measurable by either the containment hydrogen monitor or by the post accident sampling system.

A correlation has been developed which relates containment hydrogen concentration to the percent metal-water reaction for Mark I, II and III-type containments. That correlation is shown in Figure A-1*. Note A to that figure indicates the major assumptions used in developing the correlation. Note B indicates the method by which individual utilities can use the correlation to determine the extent of clad damage.

*Correlation is based on the following formula:

$$\% H_2 = \frac{(1641) \left(\frac{N}{748} \right) (MWR)}{(1641) \left(\frac{N}{748} \right) (MWR) + (3176) \left(\frac{V}{1.36 \times 10^6} \right)}$$

Where: N = number of fuel bundles
V = containment volume, ft³
MWR = fraction of cladding in active fuel zone reacting
%H₂ = concentration of hydrogen in containment atmosphere, mole %

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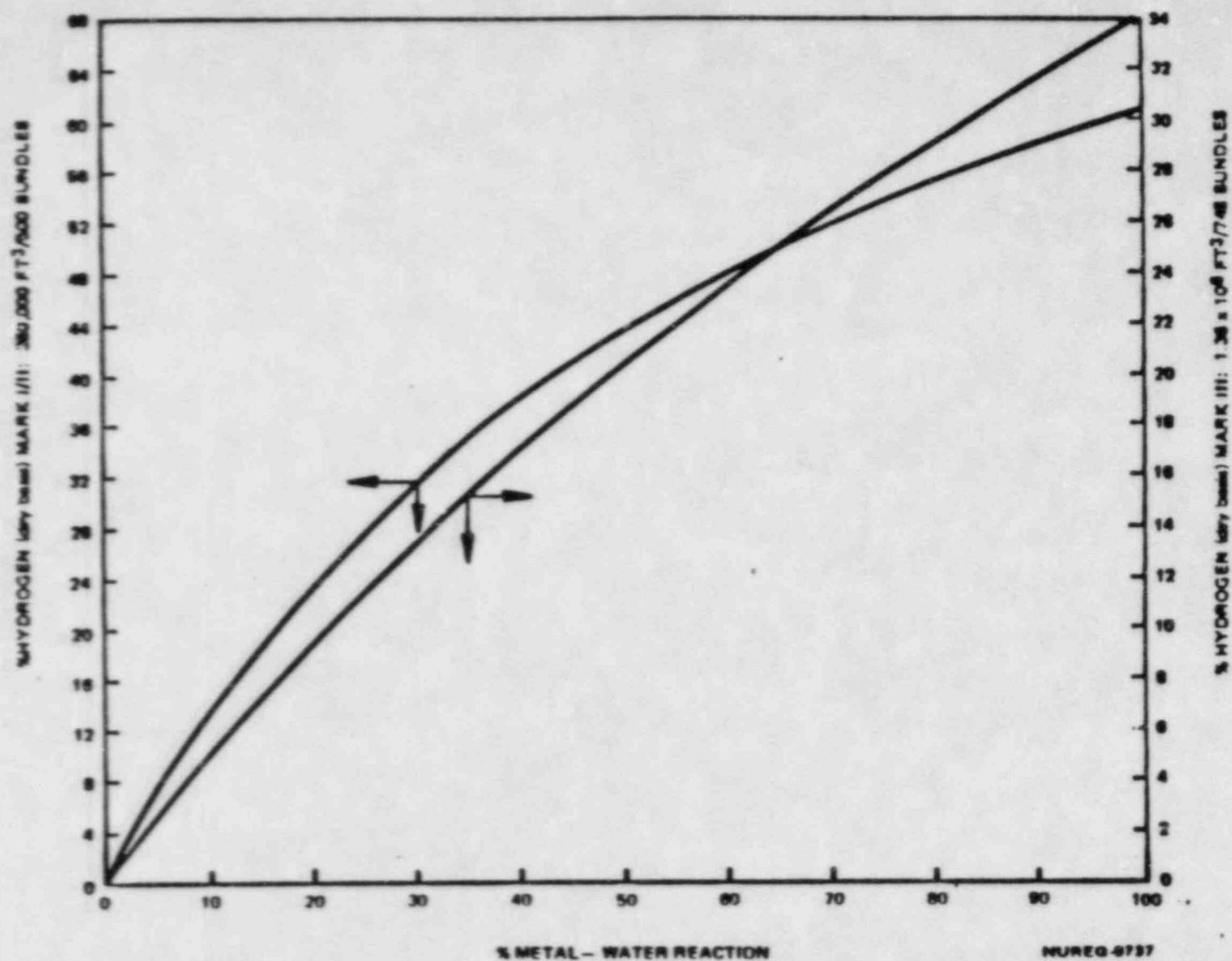


Figure A-1. Hydrogen Concentration for Mark I/II and III Containments as a Function of Metal-Water Reaction

Note A-Analytical Assumptions

1. Containment Volume = 350,000 ft³ (MK I-II)
1,360,000 ft³ (MK III)
2. Number of bundles = 500 (MK I-II)
748 (MK III)
3. Fuel type = 8x8 R
4. All hydrogen from metal-water reaction released to containment
5. Perfect mixing in containment
6. No depletion of hydrogen (e.g., containment leakage)
7. Ideal gas behavior in containment.

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Note B-Determination of Clad Damage
from Hydrogen Monitor Reading

- Step 1. Obtain containment hydrogen monitor reading, [H], in %.
- Step 2. Using the appropriate curve in Figure A-1, determine the metal-water reaction for the reference plant, MW_{ref} at [H].
- Step 3. The metal-water reaction for the actual in-plant conditions (MW) is determined from the following equation:

$$\%MW = (MW_{ref}) \left(\frac{500}{N} \right) \left(\frac{V}{350,000} \right) \quad \text{MK I-II}$$

$$\%MW = (MW_{ref}) \left(\frac{748}{N} \right) \left(\frac{V}{1,360,000} \right) \quad \text{MK III}$$

where N = number of bundles

V = total containment free volume, ft^3

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APPENDIX B

INTEGRATION OF CONTAINMENT ATMOSPHERE
RADIATION MEASUREMENT INTO CORE
DAMAGE ESTIMATE

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SUMMARY

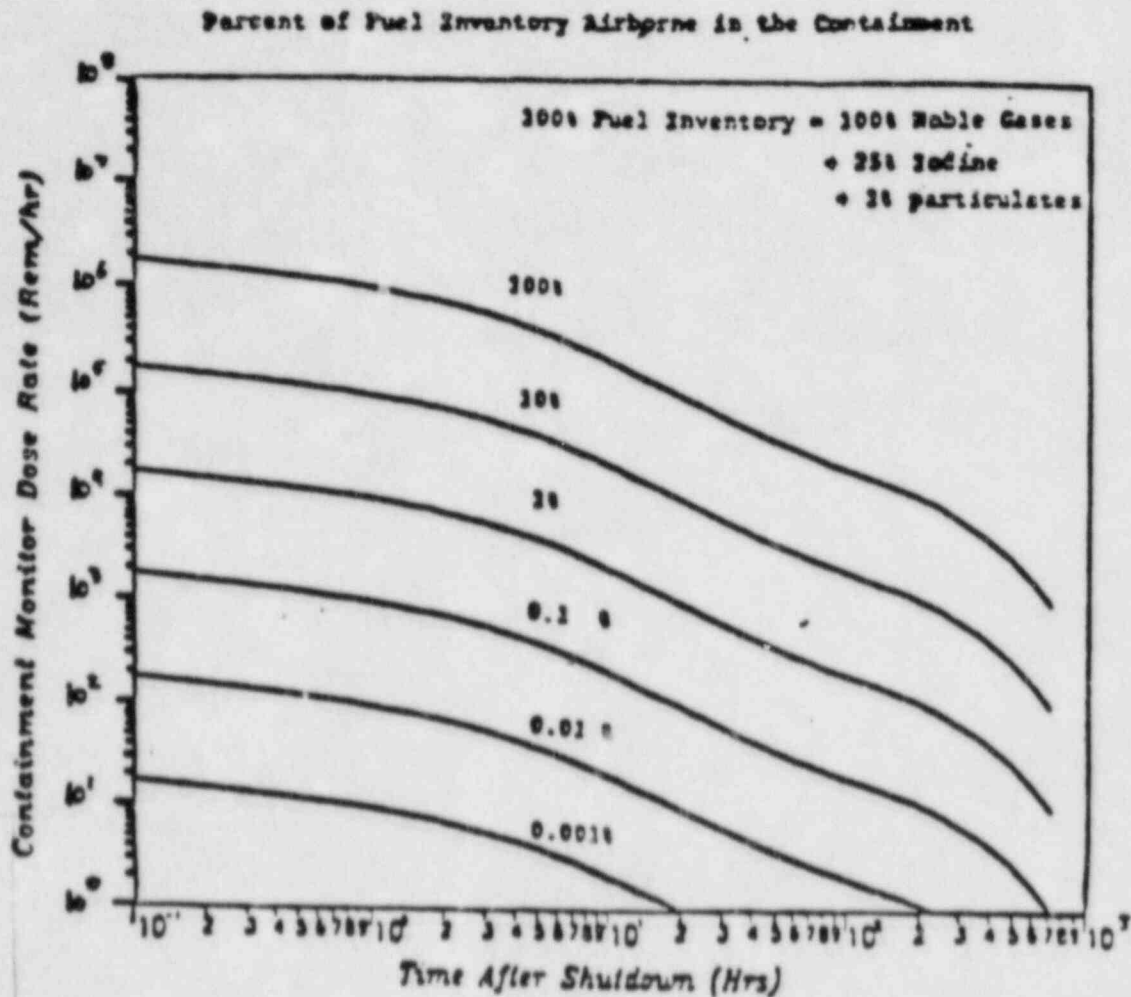
As noted in the Task 1 response, an indication of the extent of core damage is the containment radiation level which is a measure of the inventory of fission products released to the containment. Several plant procedures already contain a correlation of the containment radiation monitor dose rate to the percent of fuel inventory airborne in the containment. The purpose of this appendix is to present that correlation and provide a method whereby individual plants can use that correlation to determine the degree of core damage.

Figure B-1 provides the results of a correlation performed for the Monticello plant. The key parameters which impact the containment dose rate are reactor power, containment volume and monitor location within the containment.

The method whereby individual plants can apply this correlation is provided in Note A in Figure B-1.

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FIGURE B-1



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% Fuel Inventory Released	Approximate Source and Damage Estimate
100.	100% TID-14844, 100% fuel damage, potential core melt.
50.	50% TID noble gases, TMI source.
10.	10% TID, 100% WRC gap activity, total clad failure, partial core uncovered.
3.	3% TID, 100% WASH-1400 gap activity, major clad failure.
1.	1% TID, 10% WRC gap, Max. 10% clad failure.
.1	.1% TID, 1% WRC gap, 1% clad failure, local heating of 5-10 fuel assemblies.
.01	.01% TID, .1% WRC gap, clad failure of 3/4 fuel element (36 rods).
10 ⁻³	.01% WRC gap, clad failure of a few rods.
10 ⁻⁴	100% coolant release with spiking.
5x10 ⁻⁶	100% coolant inventory release.
10 ⁻⁶	Upper range of normal airborne noble gas activity in containment.

Note A-Determination of Clad Damage
From Containment Radiation Monitor Reading

The procedure for determination of fraction of fuel inventory released to the containment is as follows:

- Step 1: Obtain containment radiation monitor reading, [R] in Rem/hr.
- Step 2: Determine elapsed time from plant shutdown to the containment radiation monitor reading [t] in hours.
- Step 3: Using Figure B-1, determine the fuel inventory release for the reference plant $[I]_{ref}$ in %.
- Step 4: Determine the inventory release to the containment [I] using the following formula:*

$$[I] = [I]_{ref} \left(\frac{1670}{P} \right) \left(\frac{V}{237,450} \right) (6/D)$$

where P = reactor power level, Mw_{th}

V = total containment free volume, ft^3

D = distance of detector from reactor biological shield wall, ft.

*Assumes a wall-mounted detector and no major equipment in detector line-of-sight except for reactor vessel shield wall.

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ATTACHMENT D

ATTACHMENT D

TOTAL INDIVIDUAL DOSES TO OBTAIN, TRANSPORT, AND ANALYZE A MINIMUM NUMBER OF SAMPLES AT t = 1 HR

TOTALS FOR MINIMUM NUMBER OF SAMPLE PROCED.	TIME (MIN.)	WHOLE BODY (GAMMA)			EXTREMITIES (GAMMA)		
		BACKGROUND	SAMPLE	TOTAL	BACKGROUND	SAMPLE	TOTAL
		DOSE (REM)	DOSE (REM)	DOSE (REM)	DOSE (REM)	DOSE (REM)	DOSE (REM)
REF G. E. MANUAL							
1. PROCEDURE 4.3 [10 ML UNDILUTED DRYWELL ATM.]	60.	0.4484	0.6610	1.1095	0.4484	32.152	32.60
2. PROCEDURE 4.4 [IODINE CART. DRYWELL ATM.]	52.	0.4007	2.6592	3.060	0.4007	10.2403	10.642
3. PROCEDURE 4.5 [15 ML UNDILUTED DEGASSED RCS LIQUID SAMPLE]	79.	0.6032	0.2047	0.8077	0.6032	36.87	37.47
4. PROCEDURE 4.5 [15 ML DILUTED DEGASSED RCS LIQUID SAMPLE]	103.	0.6196	0.05199	0.6715	0.6196	2.2508	2.8704
5. PROCEDURE 4.6 & 4.7 [10 ML DISSOLVED GAS SAMPLE FROM DILUTED RCS LIQUID SAMPLE 10% NG]	139.	0.8630	0.1045	0.9675	0.8630	0.1715	1.0346
(*) TOTALS FOR PROCEDURES AT t = 1 HR	433. (7.22 HR)	2.935	3.681	6.616 TOTAL INDIVIDUAL WHOLE BODY DOSE AT t = 1 HR	2.935	81.685	84.617 TOTAL INDIVIDUAL EXTREMITIES DOSE AT t = 1 HR

TOTAL MAN-REM FOR SAMPLES TAKEN FOR 7 DAYS, POST-LOCA, SHOULD NOT EXCEED 11. MAN-REM W.B. and 141. MAN-REM EXTREMITIES

(*) Total of all individual samples are not required to meet the dose and time criteria specified in NUREG-0737, Section II.B.3, Post Accident Sampling Capability. The dose criteria of 5. Rem whole body and 75. Rem for Extremities, and the 3 hour time criteria, to obtain and analyze a single sample can easily be met for any sample being obtained, transported, and analyzed, above.