ATTACHMENT A

DRESDEN STATION UNITS 2 and 3

PROCEDURE FOR ESTIMATING CORE DAMAGE

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DETERMINATION OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITION

A. PURPOSE

The purpose of this procedure is to determine the extent of core damage under accident conditions.

B. REFERENCES

Procedures for the Determination of the Extent of Core Damage Under Accident Conditions, C.C. Lin, General Electric NEDO-22215.

C. PREREQUISITES

1. Post-accident High Radiation Sampling System (HRSS) or other suitable means of obtaining the necessary sampler.

2. Gamma ray spectrometry system.

D. PRECAUTIONS

Samples under accident conditions will be highly radioactive, proper radiation protection practices are to be followed when obtaining, handling, transporting or diluting highly radioactive samples.

E. LIMITATIONS AND ACTIONS

1. Measurements of Cesium (Cs)-137 and Krypton (Kr)-85 activities may be difficult to measure until most of the shorter-lived isotopes have decayed.

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

3. If water sample results show unusually high concentrations of some less volatile isotopes such as Strontium (Sr)-92, Barium (Ba)-140, Lanthanum (La)-140, and Ruthenium (Ru)-103 some degree of fuel melting may be inferred.

4. The ratio of isotopes released from either the fuel gap or the molten fuel are significantly different as shown in Table 1, thus the source (fuel or gap) of release may be identified with the use of Table 1.

5. The fission product inventories in the core are calculated based on three years (1095 days) of continuous operation at 3651 MW, or 102% of rated power for the reference plant. These parameters were used to formulate Figures 1 through 4.

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6. If the concentration of a fission product in reactor water or drywell (corrected by decay to the time of reactor shutdown), is measured to be higher than the baseline concentration shown in the lower right hand corner of Figures 1 through 4, then the extent of fuel of cladding damage can be determined from the curves in Figures 1 through 4 based on Iodine (I)-131, Cs-137, Xenon (Xe)-133, and Kr-85.

7. Sampling.

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

Event Type	Sample Location
(1) Non-Breaks (e.g., MSIV Closure)	Suppression Pool Atmosphere
(2) Small Breaks	Drywell (before depress.) Suppression Pool Atmosphere (after depress.)
(3) Large Breaks (liquid or steam) in Containment	Drywell
(4) Large Breaks outside containment	Suppression Pool Atmosphere

   b. 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 LPCI system.

   c. 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.

F. PROCEDURE

1. If not already performed, collect the required samples using the appropriate Dresden Sample Building Procedures (DSBPs).

2. Analyze the samples on a gamma ray spectrometry system using station procedures. Be sure to decay correct sample results to the proper sample time.

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3. From the data obtained in Step F.2., pick out the results for the desired fission product. Identify the fission product as:

   a. Cwi - for a fission product from a liquid sample.

   b. Cgi - for a fission product from a gas sample.

NOTE

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 (conc.) Cwi or Cgi would be averaged from the separate measurements:

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

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

4. Decay correct sample results to the time of reactor shutdown.

$e^{\lambda_i t}$ where: $\lambda_i = \frac{0.693}{t_{1/2}}$

$t_{1/2}$ = half life of fission product i

t = line from reactor shutdown to sample time

NOTE

$t_{1/2}$ and t must be in the same units of time

5. Correct the measured gaseous activity concentrations for temperature and pressure differences in the sample vial and the containment (drywell/torus) gas phase.

NOTE

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).

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

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6. Calculate the inventory correction factor (F_{Ii}). See the attached Appendix A for example of the calculation.

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

$$= \frac{3651\ (I-e^{-1095\lambda_i})}{\sum_j P_j (I-e^{-\lambda_i T_j})e^{-\lambda_i T_j^\circ}}$$

where: λ_i = decay constant of isotope i

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

T_j° = duration of operating period j (day)

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

*NOTE

In each period, the variation of steady power should be limited to ± 20%.

7. Calculate the plant parameter correction factors (Fw, Fg). Refer to Table 2.

$$Fw = \frac{\text{Operating plant coolant [grams (g)]}}{\text{Reference plant coolant mass } (3.92 \times 10^9 g)}$$

$$Fg = \frac{\text{Operating plant containment gas volume (cc)}}{\text{Reference plant containment gas volume } (4 \times 10^{10} cc)}$$

8. Calculate the reference plant equivalent concentration.

$$Cwi = (Cwi)(e^{\lambda_i t})(F_{Ii})(Fw)$$

$$Cgi = (Cgi)(e^{\lambda_i t})(F_{Ii})(Fw)$$

Cwi from Step F.3.

$e^{\lambda_i t}$ from Step F.4.

F_{Ii} from Step F.6.

Fw from Step F.7.

Cgi from Step F.5.

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9. Use Figures 1 through 4 to estimate the extent of fuel or cladding damage.

10. Determination of Clad Damage from Hydrogen Monitor Reading.

   a. Obtain containment hydrogen monitor reading, [H], in % (or use results from HRSS Samples if available).

   b. Using the appropriate curve in Figure 5, determine the metal-water reactor for the reference plant, MW_{ref} at [H].

   c. The metal-water reaction for the actual in-plant conditions (MW) is determined from the following equation:

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

   where N = number of bundles = 724 for Dresden 2 or 3

   V = total containment free volume = 2.75×10^5 ft.^{3}

11. 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:

   a. Obtain containment radiation monitor reading, [R] in Rem/hour.

   b. Determine elapsed time from plant shutdown to the containment radiation monitor reading [t] in hours.

   c. Using Figure 6, determine the fuel inventory release for the reference plant $[I]_{ref}$ in %.

   d. 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} = 2527 MW_{th}

   V = total containment free volume = 2.75×10^5 ft.^{3}

   D = distance of detector from reactor biological shield wall = 17 feet.

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12. Application of other significant parameters to core damage estimate.

a. Steps F.1. through 9. show how to determine an estimate of core damage based on radionuclide measurements. Based on these steps, 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	1	1
Cladding Failure	2	3	4
Fuel Overheat	5	6	7
Fuel Melt	8	9	10

b. 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 ten (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.

c. 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 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 overheat and fuel melt (i.e., Categories 5 through 10). Measurement of hydrogen in containment and use of the hydrogen correlation could be used to verify that extensive clad damage had occurred. Use of the containment radiation monitor reading would verify that a significant fission product release to the containment had occurred, further verifying the initial assessment.

d. 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.

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e. The flow diagram in Figure 7 indicates how the analysis of the other significant parameters relates to the estimation of core damage based on radionuclide measurements.

G. CHECKLISTS

None.

H. TECHNICAL SPECIFICATIONS REFERENCES

None.

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TABLE 1

RATIOS OF ISOTOPES IN CORE INVENTORY AND FUEL GAP

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

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

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

TABLE 2

PLANT PARAMETERS

			Primary Coolant*		Containment Gas*	
Plant	Reactor Type/ Containment Design	Rated Power (MWt)	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)
Dresden 2 & 3	BWR 3/I	2527	2.61	3.18	4.48	3.30

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

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

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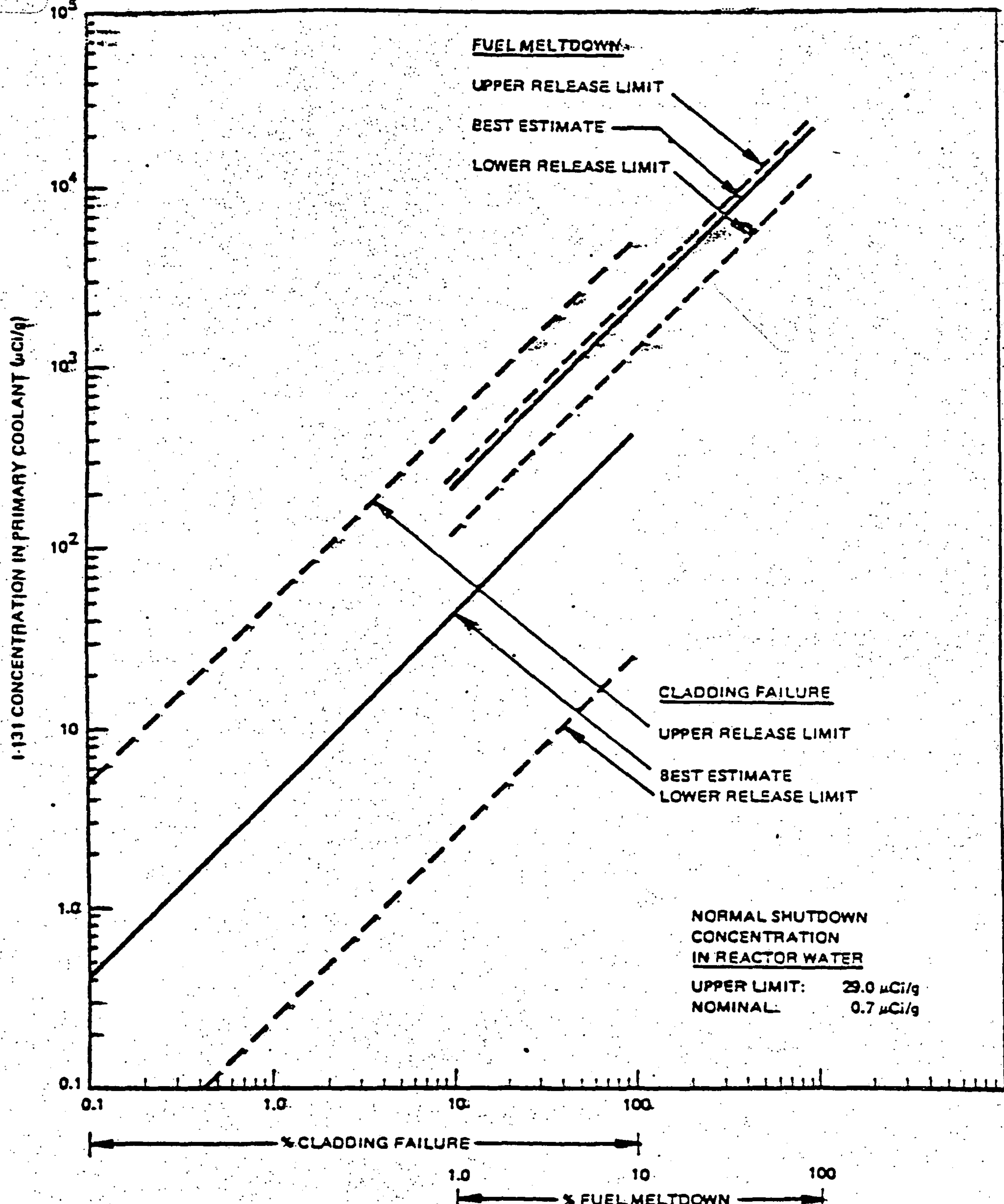

Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

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Figure 1

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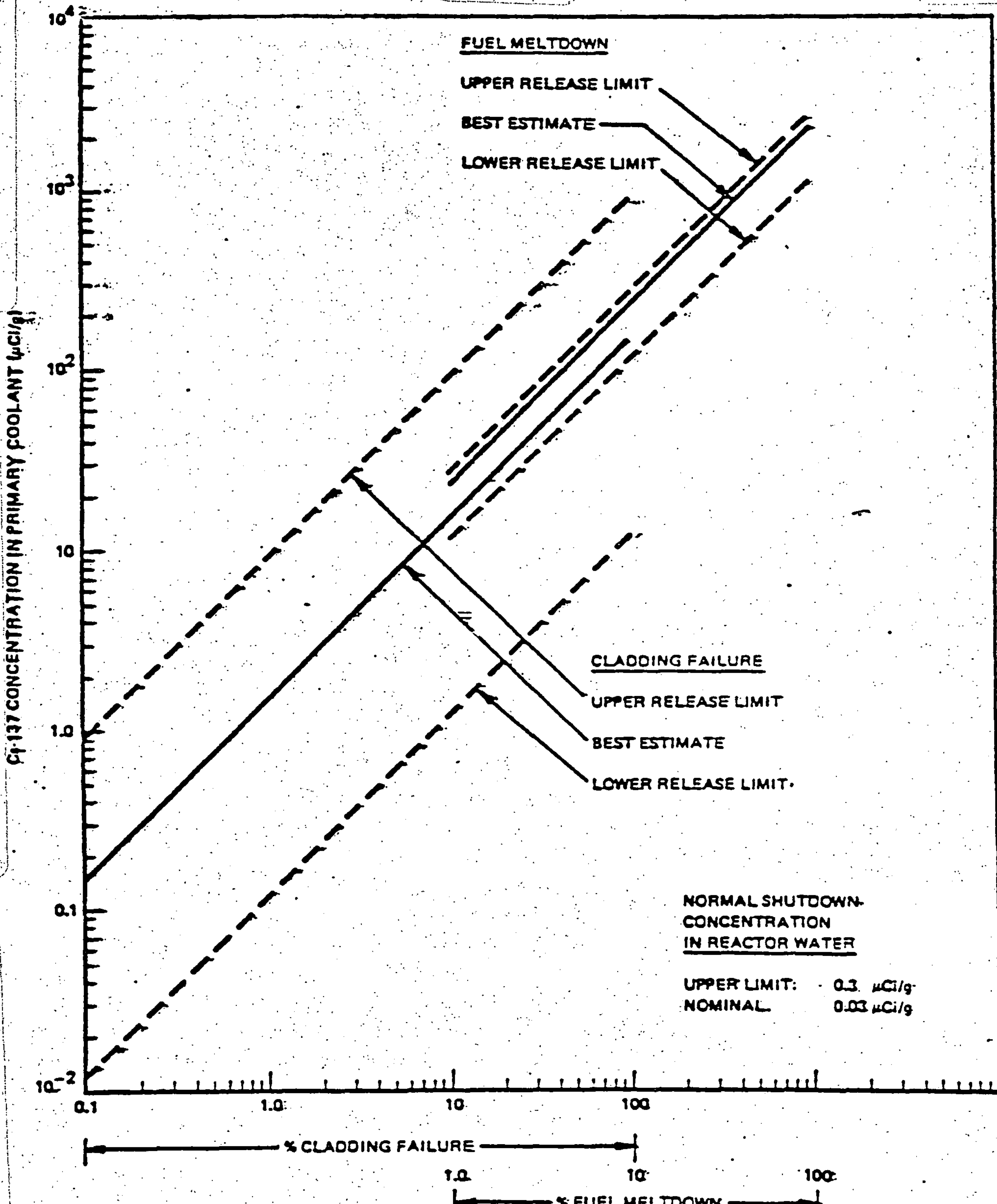

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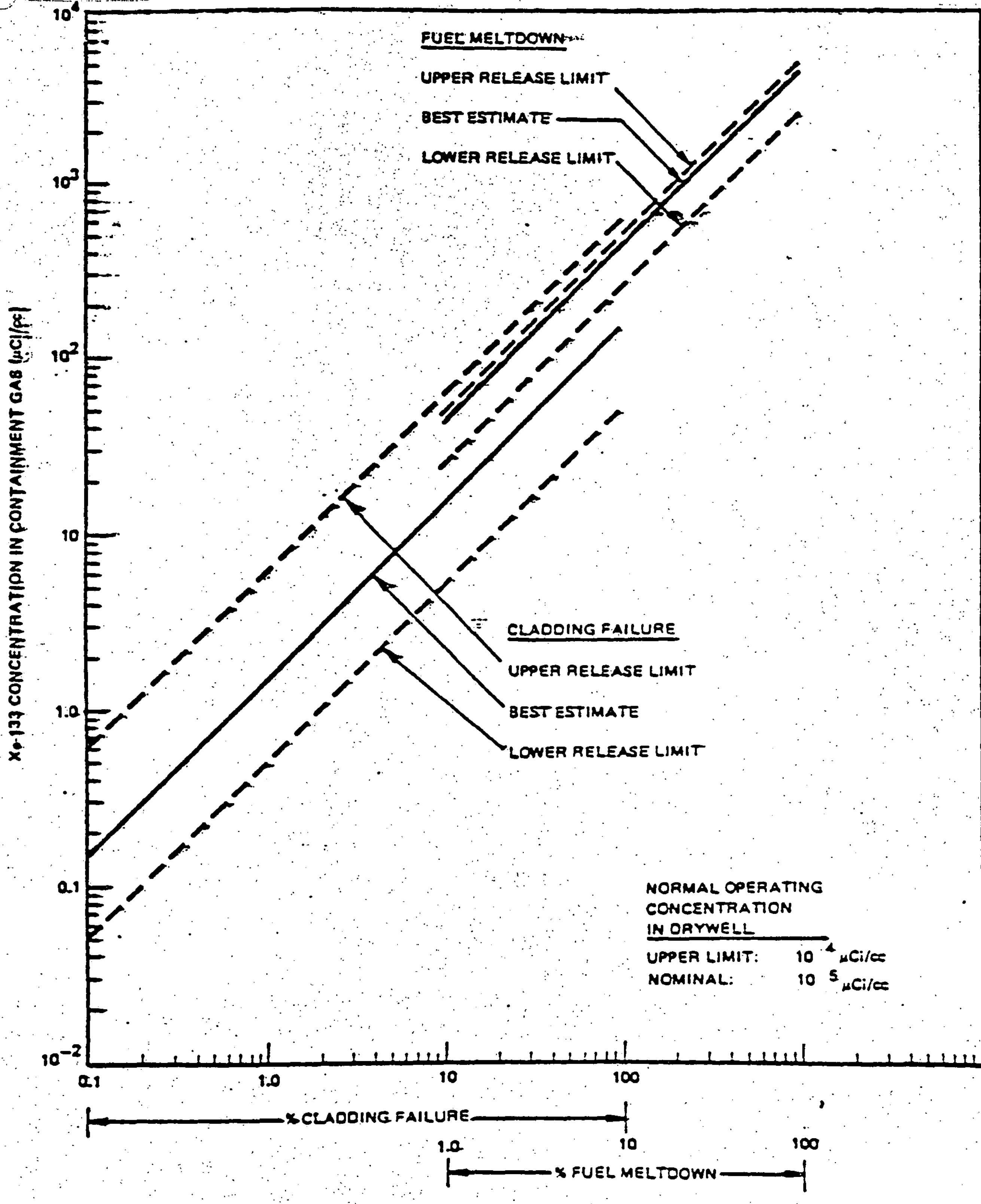
Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

Figure 2



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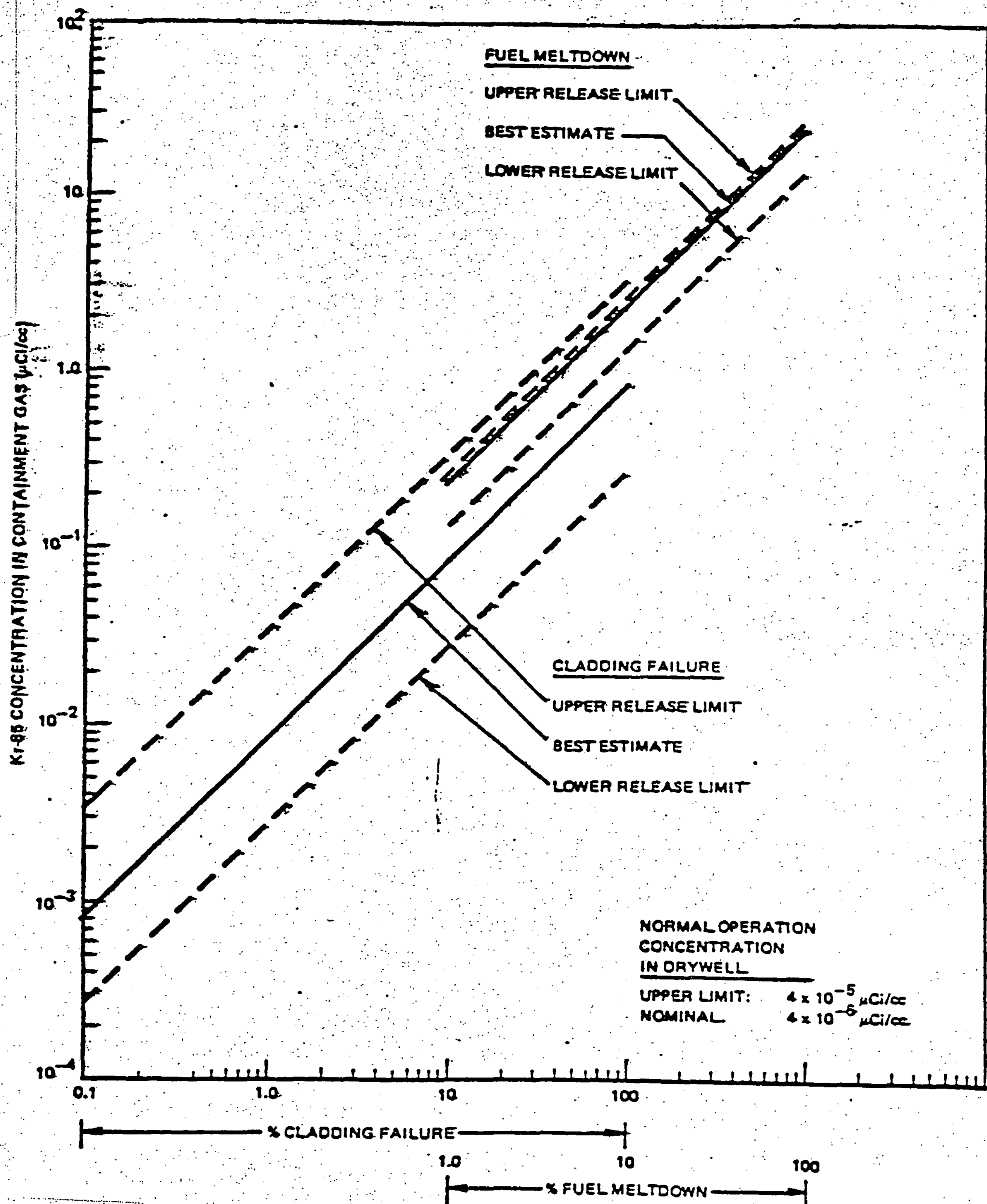
Relationship Between Xe-133 Concentration in the Containment Gas (Drywell + Torus Gas) and the Extent of Core Damage in Reference Plant

Figure 3



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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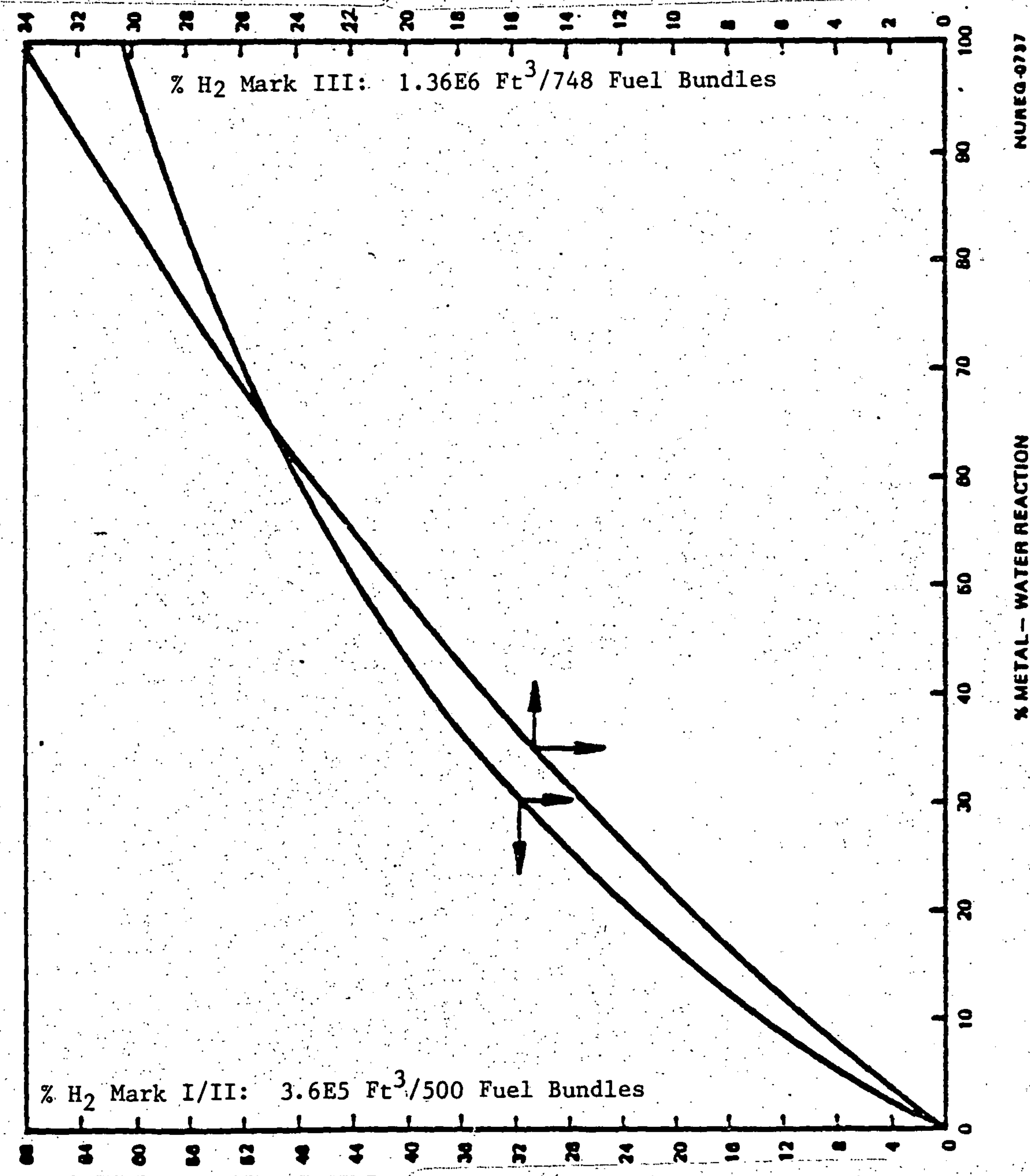
Figure 4



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

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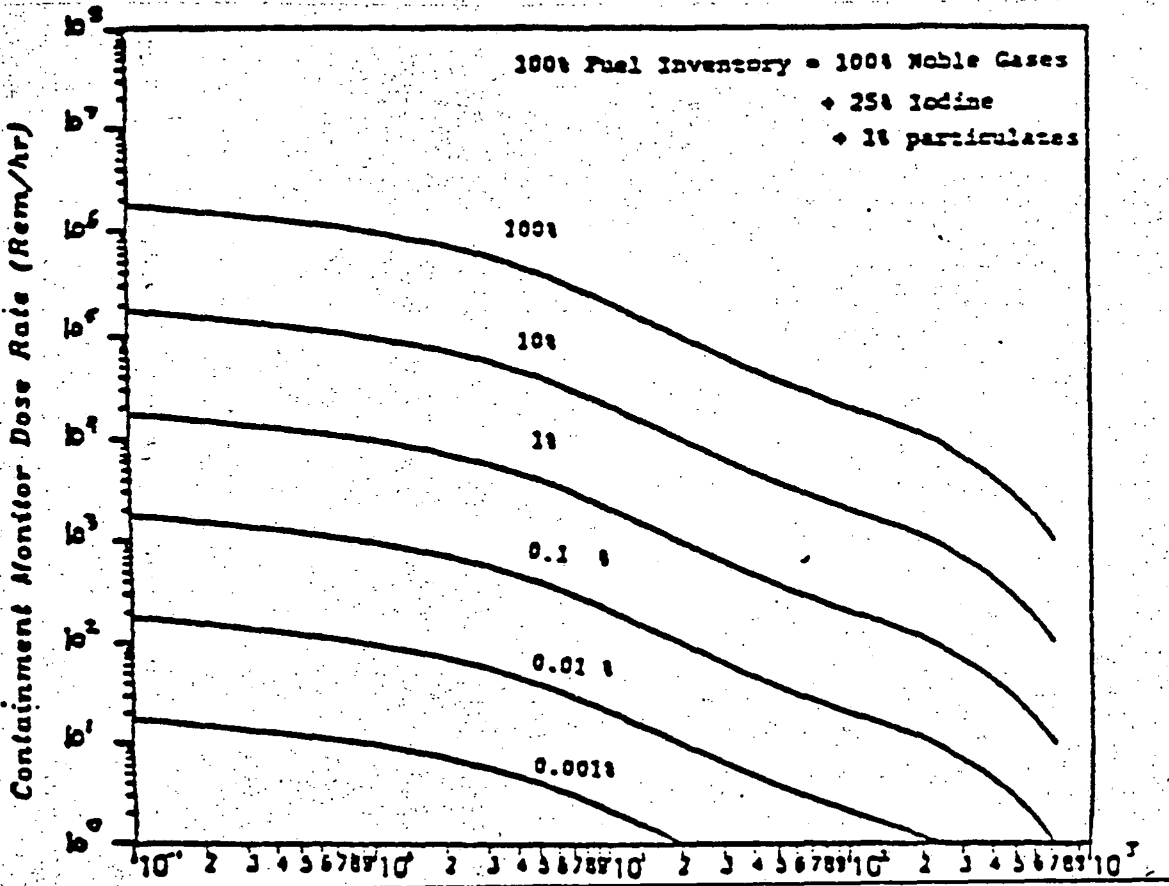
FIGURE 6

PERCENT OF FUEL INVENTORY AIRBORNE IN THE CONTAINMENT



% Fuel Inventory Released	Approximate Source and Damage Estimate
100.0	100% TID-14844, 100% fuel damage, potential core melt.
50.0	50% TID noble gases, TMI source.
10.0	10% TID, 100% NRC gap activity, total clad failure, partial core uncovered.
3.0	3% TID, 100% NRC WASH-1400 gap activity, major clad failure.
1.0	1% TID, 10% NRC gap, Max. 10% clad failure.
0.1	0.1% TID, 1% NRC gap, 1% clad failure, local heating of 5-10 fuel assemblies.
0.01	0.01% TID, 0.1% NRC gap, clad failure of 3/4 fuel element (36 rods).
10^{-3}	0.01% NRC gap, clad failure of a few rods.
10^{-4}	100% coolant release with spiking.
5×10^{-6}	100% coolant inventory release.
10^{-6}	Upper range of normal airborne noble gas activity in containment.

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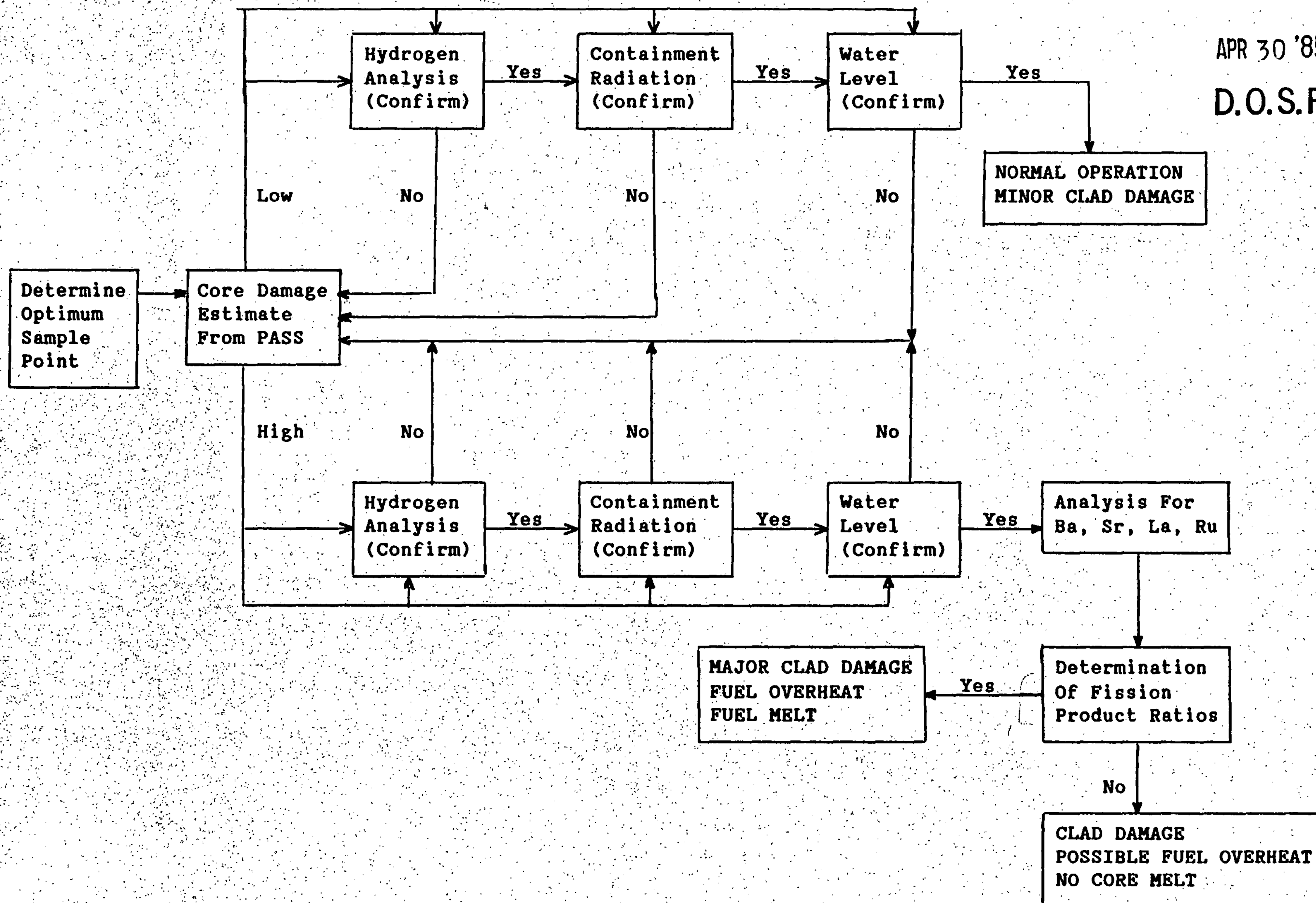
FIGURE 7
SEQUENCE OF ANALYSIS FOR
ESTIMATION OF CORE DAMAGE



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APPENDIX A

SAMPLE CALCULATION OF FISSION PRODUCT INVENTORY CORRECTION FACTOR

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

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

where:

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

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

T_j = duration of operating period j (day)

T_j° = 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°	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

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APPENDIX A (Cont'd)

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

$$F_{I(I-131)} = \frac{3651[1-e^{-0.0862(1095)}]}{1000[1-e^{-0.0862(60)}]e^{-0.0862(254)} + 2000[1-e^{-0.0862(200)}] e^{-0.0862(44)} + 3000[1-e^{-0.0862(14)}]e^{-0.0862(0)}}$$

$$= \frac{3651}{\approx 0 + 0.0225 + 2103} = 1.74$$

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

$$F_{I(Cs-137)} = \frac{3651[1-e^{-6.29E-5(1095)}]}{1000[1-e^{-6.29E-5(60)}]e^{-6.29E-5(254)} + 2000[1-e^{-6.29E-5(200)}]e^{-6.29E-5(44)} + 3000(1-e^{-6.29E-5(14)})e^{-6.29E-5(0)}}$$

$$= \frac{243.16}{3.74 + 24.93 + 2.64} = 7.77$$

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