ATTACHMENT B

QUAD CITIES STATION UNITS 1 and 2

PROCEDURE FOR ESTIMATING CORE DAMAGE

0023K

DETERMINATION OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

ID/12K

A. PURPOSE

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

B. REFERENCES

1. Procedures for the Determination of the Extent of Core Damage Under Accident Conditions, C.C. Lin., General Electric NEDO-22215.
2. QCP 1400-T3, Table of Ratios of Isotopes in Core Inventory and Fuel Gap.
3. QCP 1400-T4, Table of BWR Plant Parameters.
4. QCP 1400-T5, Relationship Between Fission Product Concentration and Core Damage.
5. QCP 1400-T6, Sample Calculation of Fission Product Inventory Correction Factor.
6. QCP 1400-T7, Percent of Fuel Inventory Airborne in the Containment.
7. QCP 1400-T8, Hydrogen Concentration for Mark I/II and III Containments as a Function of Metal-Water Reaction.
8. QCP 1400-T9, Sequence of Analysis for Estimation of Core Damage.

C. PREREQUISITES

1. Post accident high radiation sampling system (HRSS) or other suitable means of obtaining the necessary samples.
2. Gamma ray spectrometry system.
3. Containment High Range Radiation Monitoring System.
4. Containment Hydrogen Monitoring System.

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D. PRECAUTIONS

1. 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.

2. 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:

EVENT TYPE	SAMPLE LOCATION
Non Breaks (MSIV Closure)	Suppression pool atmosphere
Small Breaks	Drywell (before depressurization) Suppression pool atmosphere (after depressurization)
Large Breaks (in containment)	Drywell
(outside containment)	Suppression pool atmosphere

For liquid sampling the recommended sampling locations are as follows:

PLANT CONDITION	SAMPLE LOCATION
Recirc. System Available, Sufficient Pressure	Recirc. Loop B
Low Reactor Pressure	RHR System

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E. LIMITATIONS AND ACTIONS

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

2. If water sample results show unusually high concentrations of some less volatile isotopes such as Sr-92, Ba-140, and La-140, some degree of fuel melting may be inferred.

3. The ratio of isotopes released from either the fuel gap or the molten fuel are significantly different as shown in QCP 1400-T3, thus the source (fuel or gap) of release may be identified with the use of QCP 1400-T3.

4. 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 for the reference plant. These parameters were used to formulate the curves in QCP 1400-T5.

5. If the concentration of a fission product in reactor water or the drywell, corrected by decay to the time of reactor shutdown, is measured to be higher than the baseline concentration shown in the curves in QCP 1400-T5, then the extent of fuel or cladding damage can be determined from the curves based on I-131, Cs-137, Xe-133, and Kr-85.

6. It is recommended that both the water and gas phase samples be measured to reduce the uncertainty in core damage estimations. Confirmation of these estimations can be made by incorporating information extracted from the containment high range radiation monitoring system and the containment hydrogen monitoring system. Reactor water level should also be monitored to evaluate the extent and length of time that the core may be uncovered.

7. Significant periods of core uncovery 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.

F. PROCEDURE

1. Use of the post accident sample system to estimate core damage.

   a. If not already performed, collect the required samples using Station procedures.

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

   c. From the data obtained in step F.1.b. pick out the results for the desired fission product. Identify the fission products as:

   C_{wi} - for a fission product from a liquid sample

   C_{gi} - 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 concentration C_{wi} or C_{gi} would be averaged from the separate measurements.

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

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

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d. 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 = time from reactor shutdown to sample time

NOTE

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

e. Correct the measured gaseous activity concentration 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)

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

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f. Calculate the inventory correction factor (F_{Ii}) (See QCP 1400-T6 for a sample calculation). For a particular short-lived isotope, i, a calculation for only a period of about six half-lives of reactor operation time before reactor shutdown should be accurate enough.

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

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

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)

NOTE

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

g. Calculate the plant parameter correction factors (F_w F_g) (See QCP 1400-T4)

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

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

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h. Calculate the reference plant equivalent concentration.

$$C_{wi}^{Ref} = C_{wi} \text{ (step F.1.c.) } \times e^{\lambda_i t} \text{(step F.1.d.) } \times F_{Ii} \text{ (step F.1.f.) } \times F_w \text{ (step F.1.g.)}$$

$$C_{gi}^{Ref} = C_{gi} \text{ (step F.1.e.) } \times e^{\lambda_i t} \text{(step F.1.d.) } \times F_{Ii} \text{ (step F.1.f.) } \times F_g \text{ (step F.1.g.)}$$

i. Use the curves in QCP 1400-T5 to estimate the extent of fuel or cladding damage.

2. Use of the containment radiation monitoring system to estimate core damage.

NOTE

Contaminent radiation monitor readings can provide an indication of the extent of core damage by providing a measure of the inventory of fission products released to the containment.

a. Obtain containment radiation monitor reading; [R] in rem/hr.

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

c. Using QCP 1400-T7, determine the fuel inventory release for the reference plant $[I]_{ref}$ in percent.

d. Determine the inventory release to the containment [I] using the following:

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

where P = reactor power level, MWt (2511 MWt)

V = total containment free volume, ft^3 (275,000 ft^3)

D = distance of detector from reactor biological shield wall, ft (Unit 1: A = 13.75 ft, B = 17 ft; Unit 2: A=17 ft, B = 13.75 ft)

3. Use of the containment hydrogen monitoring system to estimate core damage.

NOTE

Containment hydrogen concentration can provide an indication of the extent of core damage by providing a measure of the metal-water reaction. That concentration is measured by either the containment hydrogen monitor or by the post-accident sampling system.

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a. Obtain containment hydrogen monitor reading, [H], in percent.

b. Using QCP 1400-T8 determine the metal-water reaction for the reference plant, MW_{ref} at [H].

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

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

where N = number of bundles (724)
V = total containment volume, ft^3 (275,000 ft^3)

4. Evaluation of all plant parameters to determine core damage.

NOTE

The objective of the core damage procedure is to narrow down, to the maximum extent possible, those categories which apply to the actual in-plant situation. 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

Conditions of more than one category could exist simultaneously.

a. Use all the information available from performing the previous analyses and, with QCP 1400-T9 as a guide, estimate the extent and type of core damage.

G. CHECKLISTS

1. None.

H. TECHNICAL SPECIFICATION REFERENCES

1. None.

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Tables of

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 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*

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

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

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TABLE OF BWR

PLANT PARAMETERS

Plant	Reactor Type/ Containment Design	Rated Power (MWt)	Primary Coolant*: Reactor Water Mass (10^8 g)	Primary Coolant*: Suppression Pool Water (10^9 g)	Containment Gas*: Drywell Gas Volume (10^9 cc)	Containment Gas*: 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
Quad Cities ½	BWR 3/I	2511	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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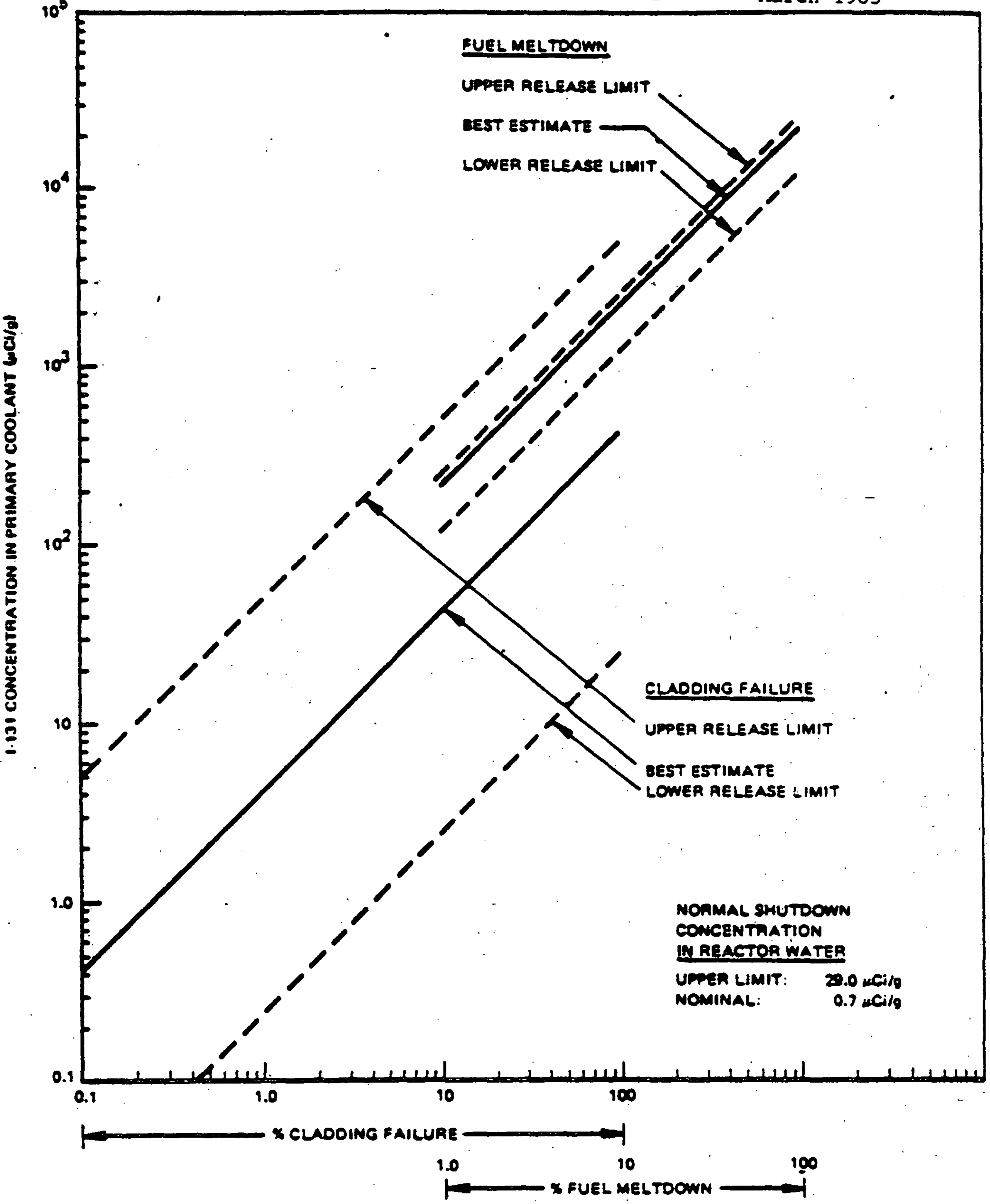
RELATIONSHIP BETWEEN FISSION PRODUCT
CONCENTRATION AND CORE DAMAGE



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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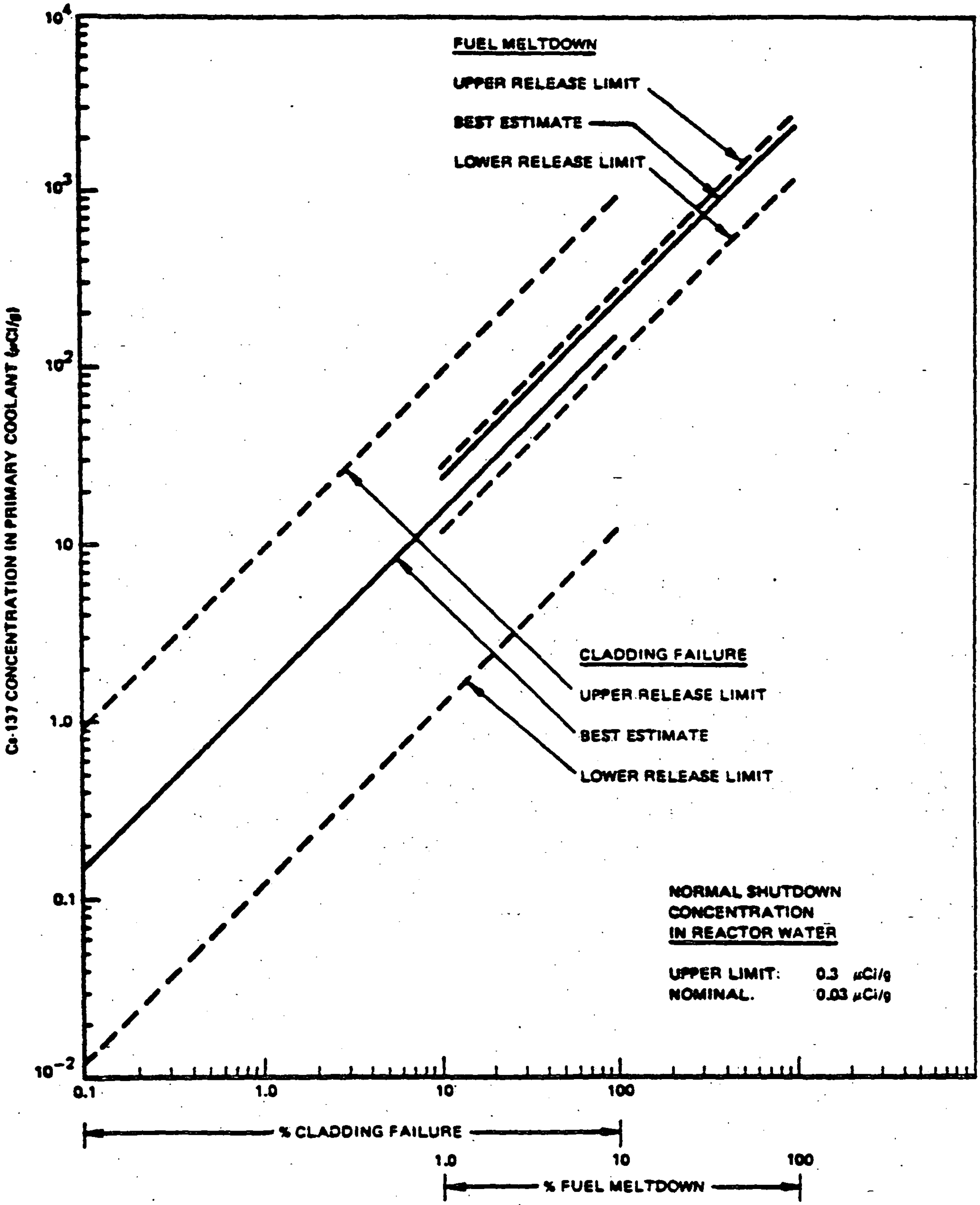
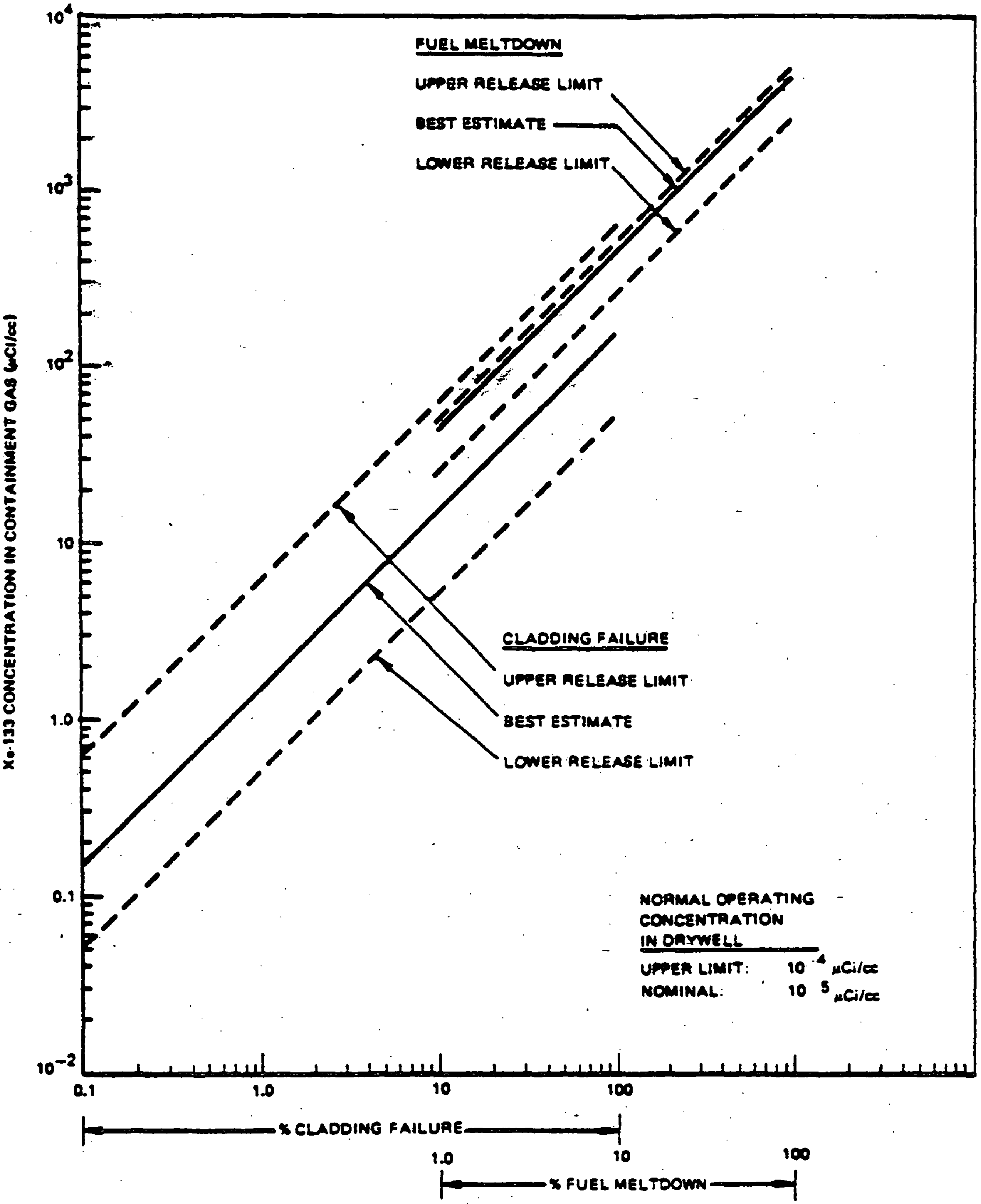


Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant



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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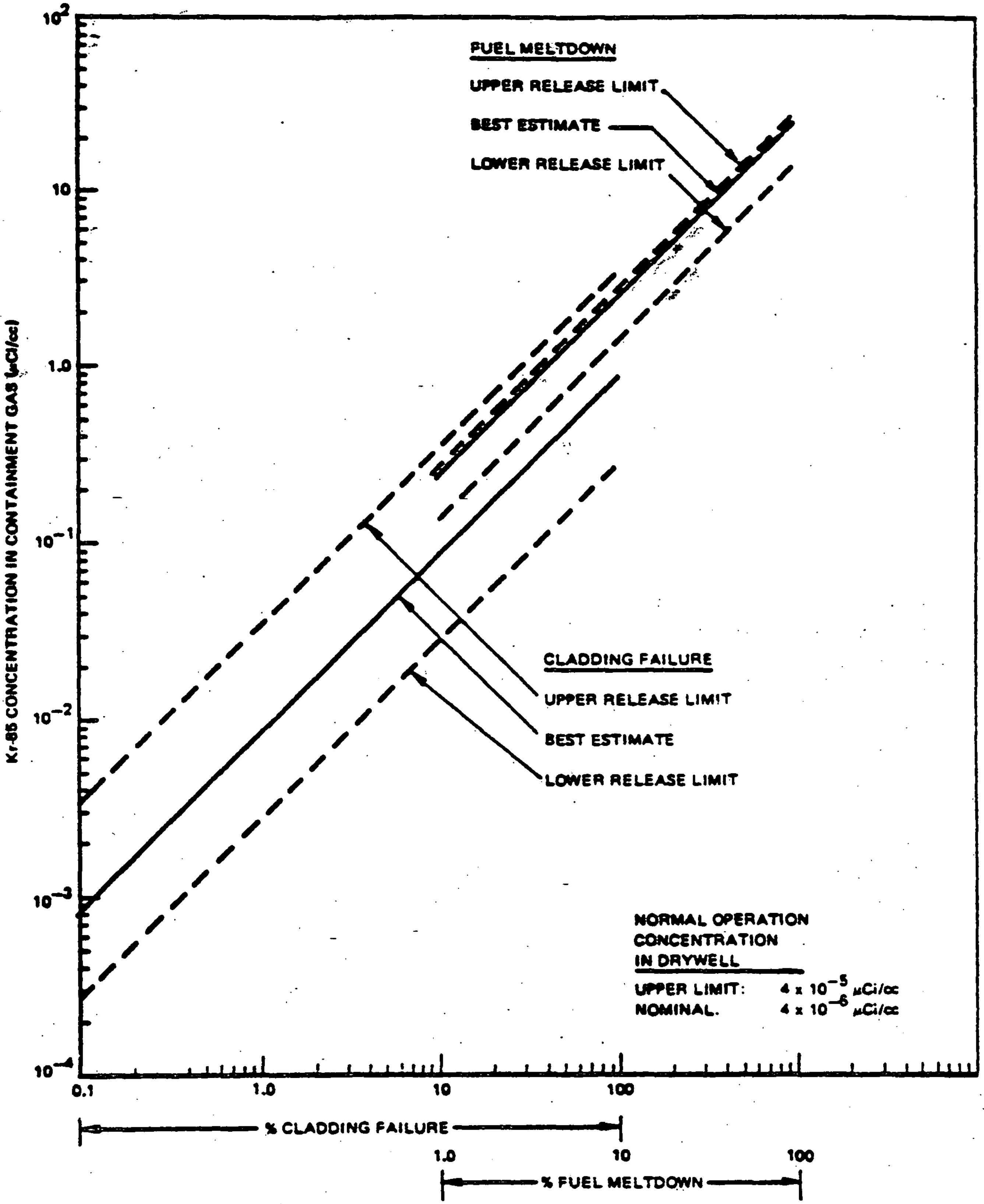

Relationship Between Kr-85 Concentration in the Containment Gas (Drywell + Torus Gas) and the Extent of Core Damage in Reference Plant

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SAMPLE CALCULATION OF FISSION PRODUCT INVENTORY CORRECTION FACTORS

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

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

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	200	44	2000
2B	271 - 300	—	—	0
3	301 - 314	14	0	3000

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

$$F_{I(I-131)} = \frac{3651(1-e^{-0.0862x1095})}{1000(1-e^{-0.0862x60})e^{-0.0862x254} + 2000(1-e^{-0.0862x200})e^{-0.0862x44} + 3000(1-e^{-0.0862x14})e^{-0.0862x0}}$$

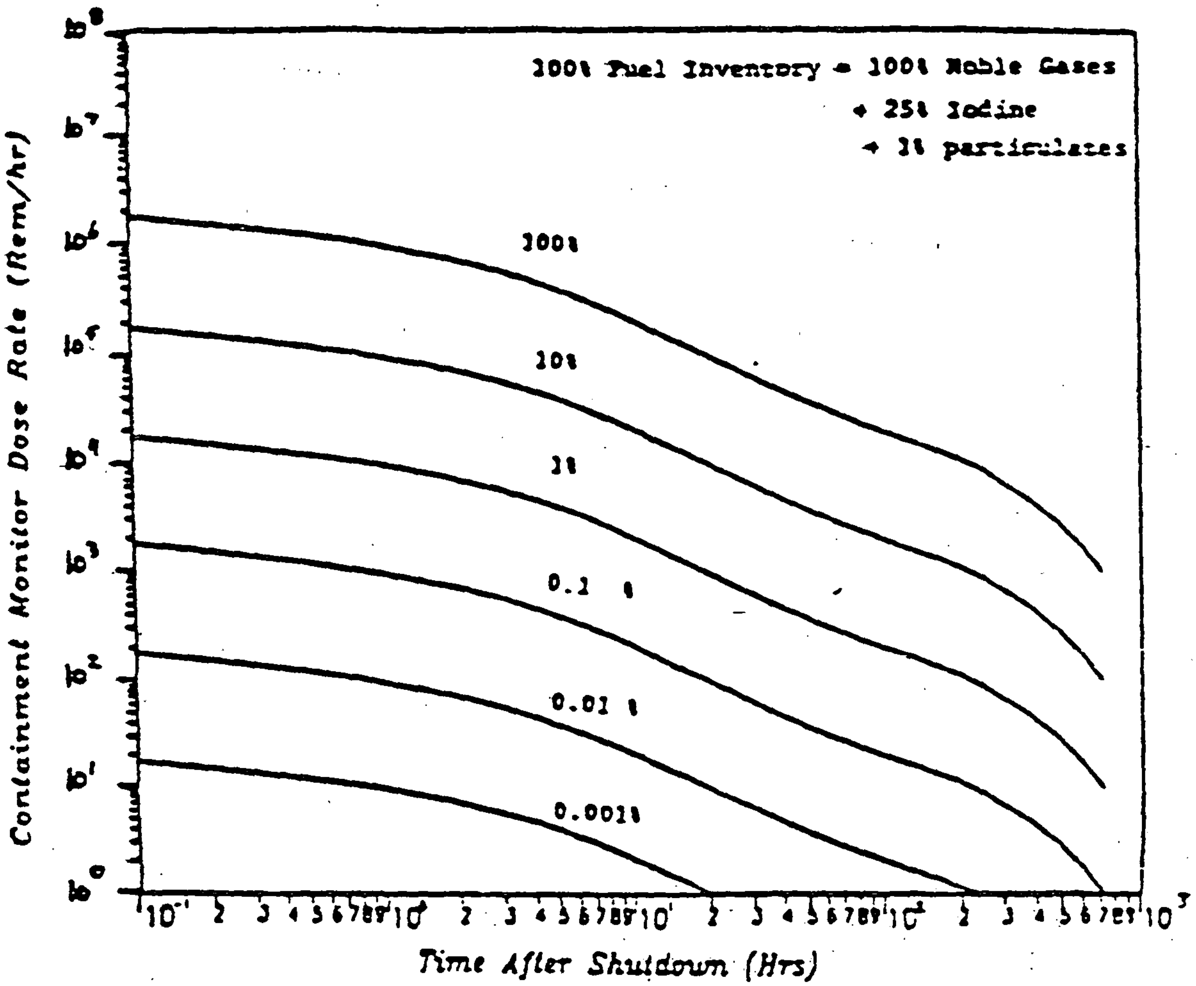
$$= \frac{3651}{\sim 0 + .45 + 2103} = 1.7$$

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

$$F_{I(Cs-137)} = \frac{3651(1-e^{-6.29x10^{-5}x1095})}{1000(1-e^{-6.29x10^{-5}x60})e^{-6.29x10^{-5}x254} + 2000(1-e^{-6.29x10^{-5}x200})e^{-6.29x10^{-5}x44} + 3000(1-e^{-6.29x10^{-5}x14})e^{-6.29x10^{-5}x0}}$$

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

Percent of Fuel Inventory Airborne in the Containment



% 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% NRC gap activity, total clad failure, partial core uncovered.
3.	3% TID, 100% WASH-1400 gap activity, major clad failure.
1.	1% TID, 10% NRC gap, Max. 10% clad failure.
.1	.1% TID, 1% NRC gap, 1% clad failure, local heating of 5-10 fuel assemblies.
.01	.01% TID, .1% NRC gap, clad failure of 3/4 fuel element (36 rods).
10^{-3}	.01% NRC 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.

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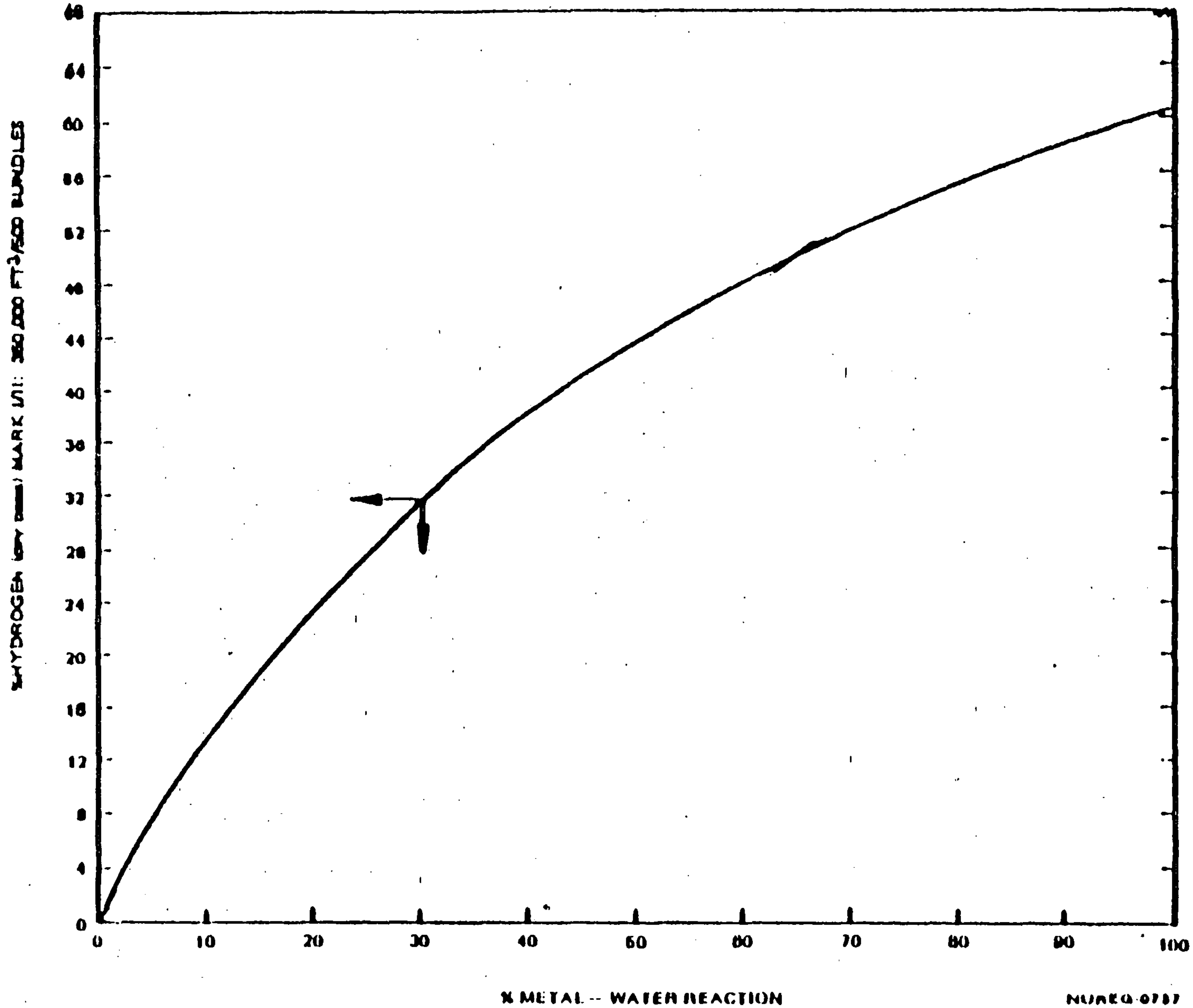

Hydrogen Concentration for Mark I/II and III Containments as a Function of Metal-Water Reaction

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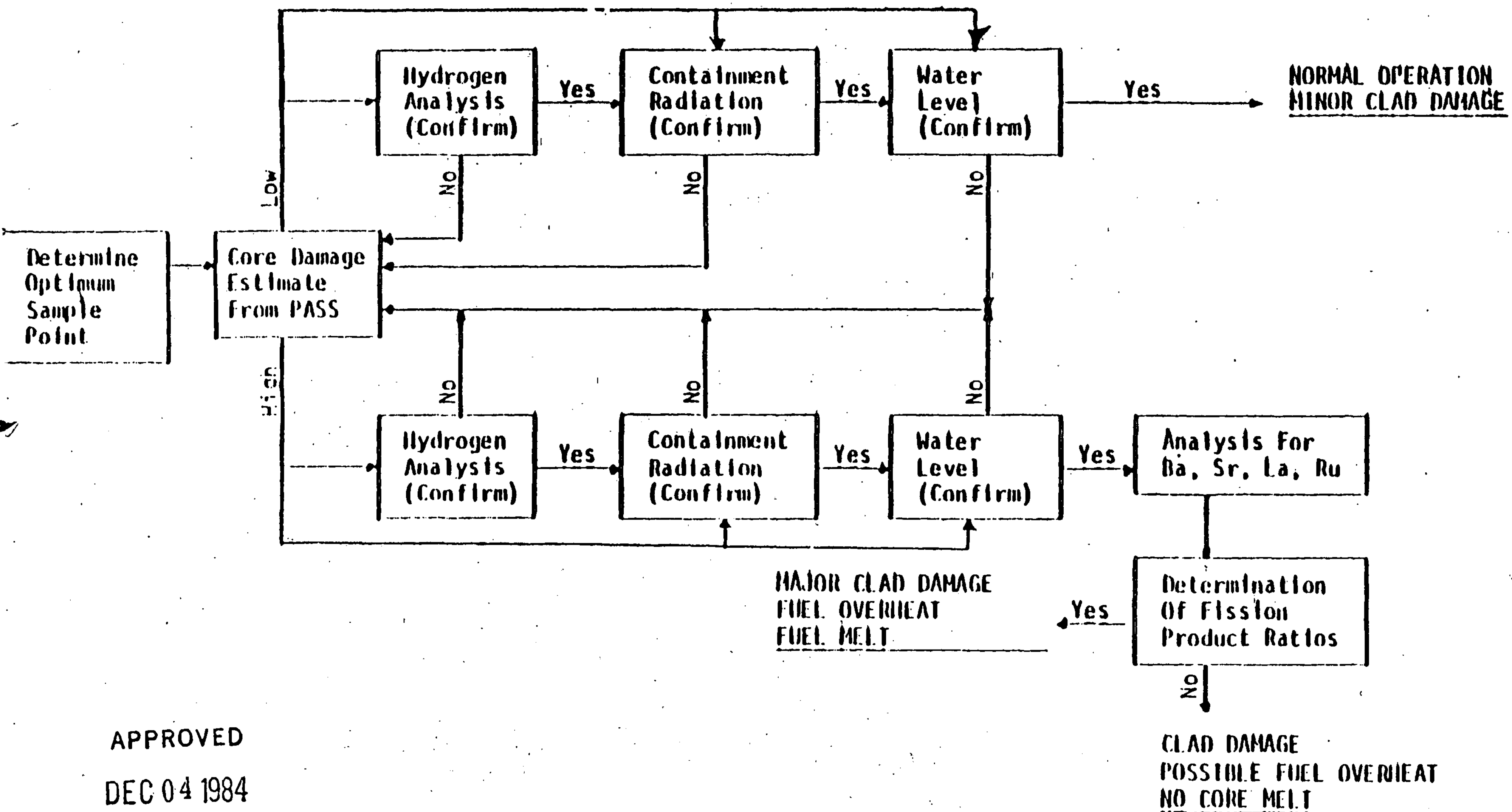
SEQUENCE OF ANALYSIS FOR ESTIMATION OF CORE DAMAGE



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