Attachment 2
Radiation Chemistry Procedure 78.000.15, Rev. 1
"Determination of Extent of Core Damage"

SR
Safety-Designation

ENRICO FERMI ATOMIC POWER PLANT UNIT 2 PCM PROCEDURE - Radchem

TITLE: Determination of Extent of Core Damage

PROCEDURE NUMBER: 78.000.15

REVISION NUMBER: 1

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***Signatures required for Safety-Related or Superintendent-Designated procedures.**

[redacted] EKS

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1.0 Purpose

The purpose of this procedure is to provide a guide for estimating the amount and type of reactor core damage under accident conditions.

2.0 Discussion

The estimation of core damage is based on the isotopic analysis of water and/or gas samples from the primary system and the integration of these results with other known plant parameters. The following matrix may be used in the assessment of core damage:

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 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, Enclosure 11 could be used to verify that extensive clad damage had occurred. Use of the containment radiation monitor reading along with the correlation provided in Enclosure 12 of this procedure would verify that a significant fission product release to the containment had occurred, further verifying the intital assessment.

The isotopic estimation of core damage will be calculated by comparing the measured concentrations of major fission products in either gas or liquid samples, after appropriate normalization, with reference plant data from a BWR-6/238 with a Mark III containment. Fission product inventories in the primary system were calculated based on postulated loss of coolant accident (LOCA) conditions after 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. The inventories of major fission products in the core at the time of reactor shutdown are given in Enclosure 1.

The pertinent reference and EF2 plant parameters are given below:

	Reference Plant	EF2
Rated Reactor Thermal Power	3579 MWt	3292 MWt
Number of Fuel Bundles	748 Bundles	764 Bundles
Total Primary Coolant Mass (Reactor Water plus Suppression Pool water)	3.92×10^9g	3.51×10^9g
Total Containment and Drywell Gas Space Volume	4.0×10^{10}cc	8.35×10^9cc

Gas/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, decay corrected to the time of reactor shutdown, is measured to be higher than the baseline concentration shown in Enclosure 2, the extent of fuel or cladding damage can be determined directly from Enclosures 5-8 based on isotopes I-131, Cs-137, Xe-133, and Kr-85. Measurements of Cs-137 and Kr-85 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.

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

NOTE: The fuel gap fission products are assumed to be released upon the rupture of fuel cladding; the majority of fission products in the core will be released when the fuel is melted at higher temperatures.

For example, if less volatile fission products such as isotopes of Sr, Ba, La, and Ru are found to have unusually high concentrations in the water sample as compared to baseline reactor water concentrations, a fuel meltdown may be assumed. The presence of 2.7hr Sr-92 (1.385MeV) and 40 hour La-140 (1.597MeV) will be relatively easy to identify and measure from a gamma ray spectrum.

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 Enclosure 3, thus the source (fuel or gap) of release may be identified.

- Samples acquired for the estimation of core damage shall be taken from locations that are consistent with break case and system conditions (Enclosure 10). This will ensure the viability of results reported and provide the best isotopic estimation of core damage.

Correlations similar to those which are 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 indicating the possibility 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 uncovery, 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.

Enclosure 13 indicates how the analysis of these other significant parameters relates to the estimation of core damage based on radionuclide measurements.

3.0 References

*3.1 Plant Operations Manual (POM) Procedure 78.000.14 (Post Accident Sampling and Analysis)

*3.2 POM Procedure 76.000.05 (Operation of the Chemistry ND6685) - (later)

*Denotes "Use" Reference

*3.3 POM Procedure 76.000.06 (Operation of the Chemistry ND680)

3.4 Lin, Chien C, Procedures for the Determination of Core Damage Under Accident Conditions, General Electric Co., 1982

4.0 Equipment Required

4.1 Apparatus - Gamma Spectroscopy system.

4.2 Reagents - None

5.0 Precautions and Limitations

None

6.0 Prerequisites

6.1 Accident conditions exist and a decision has been made to take a sample by the General Supervisor of Chemistry or designee.

6.2 Specific location and additional instructions for the acquisition of samples have been given to Operations and Chemistry consistent with the break case and system conditions as described in Enclosure 10.

7.0 Procedure

7.1 Estimation Procedure

7.1.1 Obtain samples, consistent with Enclosure 10, from the Post Accident Sampling System per Reference 3.1.

7.1.2 Perform gamma spectroscopy (per References 3.2 and 3.3) and determine the concentration of fission products I-131, Cs-137, Xe-133, and Kr-85. (Cwi in water or Cgi in gas.)

NOTE: Measurements of Cs-137 and Kr-85 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.

7.1.3 If the temperature and pressure of the gas sample vial are different from that in the containment, correct the measured gaseous activity concentration for temperature and pressure per paragraph 7.3.

*Denotes "Use" Reference

7.1.4 Calculate the fission product inventory correction factor F_{Ii} per paragraph 7.4.

7.1.5 Calculate the plant parameter correction factors (Fg or Fw) per paragraph 7.5.

7.1.6 Calculate the normalized concentration, C_{wi}^{Ref} or or C_{gi}^{Ref} for I-131, Cs-137, Xe-133, and Kr-85 per paragraph 7.6

7.1.7 Interpretation of C_{gi}^{Ref} or C_{wi}^{Ref}

1. If the normalized concentrations, C_{wi}^{Ref} or C_{gi}^{Ref}, obtained in paragraph 7.1.7 are higher than the baseline concentrations shown in Enclosure 2, the extent of fuel or cladding damage can be determined directly from Enclosures 5-8.

2. If the normalized concentrations fall into a range where release of the fission product from the fuel gap or the molten fuel cannot be definitely determined, the presence of Sr, Ba, La and Ru should be established. Fission products 2.7hr Sr-92 (1.385 MeV) and 40hr La-140 (1.597MeV) are relatively easy to identify and measure from a gamma ray spectrum and are indicative of fuel meltdown. These results should be compared to baseline reactor water concentrations.

7.2 Identification of Release Source

7.2.1 Determine the concentrations of the following short-lived isotopes by gamma spectroscopy:

Kr-87	I-134
Kr-88	I-132
Kr-85m	I-135
Xe-133	I-133
	I-131

7.2.2 Correct the measured fission products to the time of reactor shutdown.

7.2.3 Calculate isotopic ratio's per paragraph 7.7.

7.2.4 Determine release source by comparing results obtained in paragraph 7.2.3 to ratio's supplied in Enclosure 3.

7.3 Temperature and pressure correction for gas sample vial.

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

where

C_{gi} (vial) = Sample vial isotopic concentration

C_{gi} = Containment isotopic concentration

(P_1, T_1) = Sample vial pressure and temperature

(P_2, T_2) = Containment pressure and temparature

7.4 Fission Product Inventory Correction Factor

$$F_{Ii} = \frac{\text{Inventory in reference plant}}{\text{Inventory in EF2}}$$

$$= \frac{3651\ (1-e^{-\lambda i\ 1095})}{\sum_j [P_j (1-e^{-\lambda T_j})^{-\lambda i\ T_j^o}]}$$

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

T_j = duration of operating period j (days)*

T_j^o = time between the end of operating period j and time of reactor shutdown (days)*

λi = decay constant for a particular isotope (days $^{-1}$).

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. It should be pointed out that the computer calculation of core inventory takes into account the fuel burnup, plutonium fission and neutron capture reactions. 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 (Enclosure 9) and other assumptions.

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

7.5 Plant Parameter Correction Factors

$$F_w = \frac{\text{EF2 coolant mass (g)}}{\text{reference plant coolant mass } (3.92\text{x}10^9 \text{ g})}$$

$$F_g = \frac{\text{EF2 gas volume (cc)}}{\text{reference plant containment gas vol. } (4\text{x}10^{10} \text{ cc})}$$

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:

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

$$C_{gi} = \frac{\text{(Conc. in drywell)(Drywell gas vol) + (Conc. in Torus)(Torus gas vol)}}{\text{Drywell gas volume + Torus gas volume}}$$

7.6 Calculation of Normalized Concentration C_{wi} and C_{gi}

$$C_{wi}^{Ref} = C_{wi}e^{\lambda i t} \times F_{Ii} \times F_w$$

$$C_{gi}^{Ref} = C_{gi}e^{\lambda i t} \times F_{Ii} \times F_g$$

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

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

C_{wi} = measured concentration of isotope i in EF2 coolant at time, t (Ci/g)

C_{gi} = measured concentration of isotope i in EF2 containment gas at time, t (Ci/cc)

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

λi = decay constant of isotope i (day)

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

F_{Ii} = inventory correction factor for isotope i

F_g = containment gas volume correction factor

F_w = primary coolant mass correction factor

7.7 Calculation of Isotopic ratios

$$\text{Noble gas ratio} = \frac{\text{Noble gas isotopic concentration}}{\text{Xe-133 Concentration}}$$

$$\text{Iodine ratio} = \frac{\text{Iodine isotopic concentration}}{\text{I-131 Concentration}}$$

7.8 Metal-Water Reaction

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 develoepd which relates containment hydrogen concentration to the percent metal-water reaction for Mark I type containments. That correlation is shown in Enclosure 11*. Note A to that Enclosure indicates the major assumptions used in developing the correlation. 7.8.1 through 7.8.3 details the method to determine the extent of clad damage.

7.8.1 Obtain containment hydrogen reading, [H], in %.

7.8.2 Using the curve on page 1 of Enclosure 11, determine the metal-water reaction for the reference plant, MW_{ref} at [H].

7.8.3 The metal-water reaction for EF2 (%MW) is determined from the following equation:

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

N = number of fuel bundles at EF2 = 764
V = total containment free volume at EF2 in cubic feet = 292,000

7.9 Containment Radiation Level

Another indicator 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. Enclosure 12 provides the

*Correlation is based on the following formula:

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

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. To apply this correlation to EF2 perform the following:

7.9.1 Obtain containment radiation monitor reading, [R] in Rem/hr.

NOTE: Reading should be taken from CHRRM D11-N443A. Readings taken from D11-N443B may be in error (high or low) due to line B31-06-H-40-B13 blocking the detectors line-of-sight to the shield wall.

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

7.9.3 Using Enclosure 12 determine the fuel inventory release for the reference plant $[I]_{ref}$ in %.

7.9.4 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 = 292,000
D = distance of detector from reactor biological shield wall, ft. = 6.5' for D11-N443A and 6' for D11-N443B.

7.10 Refer to Enclosure 13 for estimating core damage.

8.0 Acceptance Criteria

NA

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-85m	4.48h	24.6	151(0.755)
	Kr-85	10.72y	1.1	514(0.0043)
	Kr-87	76. m	47.1	403(0.494)
	Kr-88	2.84h	66.8	196(0.203),1530(0.109)
	Xe-133	5.25d	202.	81(0.371)
	Xe-135	9.09h	26.1	250(0.906)
Halogens	I-131	8.04d	96.	364(0.824)
	I-132	2.29h	140	668(0.99),773(0.762)
	I-133	20.8 h	201	530(0.87)
	I-134	52.6 m	221	847(0.954),884(0.653)
	I-135	6.59h	189	1132(0.231),1260(0.293)
Alkali Metals	Cs-134	2.06y	19.6	605(0.98),796(0.88)
	Cs-137	30.17y	12.1	662(0.85)
	Cs-138	32.2 m	2990.**	463(0.267),1436(0.75)
Tellurium Group	Te-132	78. h	138	228(0.88)
Noble Metals	Mo-99	66.02h	183	740(0.138)
	Ru-103	39.4 d	155	497(0.9)
Alkaline Earths	Sr-91	9.52h	115	750(0.24)
	Sr-92	2.71h	123	1385(0.9)
	Ba-140	12.8 d	173	537(0.238)
Rare Earths	Y-92	58.6 d	118	934(0.137)
	La-140	40.2 h	184	487(0.453),1597(0.953)
	Ce-141	32.5 d	161	145(0.49)
	Ce-144	284.4 d	129	134(0.108)
Refractories	Zr-95	46. d	161	724(0.435),757(0.543)
	Zr-97	16.8 h	166	743(0.933)

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

**1 hr after shutdown

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

ISOTOPE	REACTOR WATER, uCi/g		DRYWELL GAS uCi/cc	
-	UPPER LIMIT	NOMINAL	UPPER LIMIT	NOMINAL
I-131	29	0.7	---	---
Cs-137	0.3*	0.03**	---	---
Xe-133	---	---	10^{-4}*	10^{-5}**
Kr-85	---	---	$4x10^{-5}$*	$4x10^{-6}$**

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

**Assuming 10% of the upper limit values.

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 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	2.28h	1.46	0.127
I-135	6.59h	1.97	0.364
I-133	20.8 h	2.09	0.685
I-131	8.04d	1.0*	1.0*

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

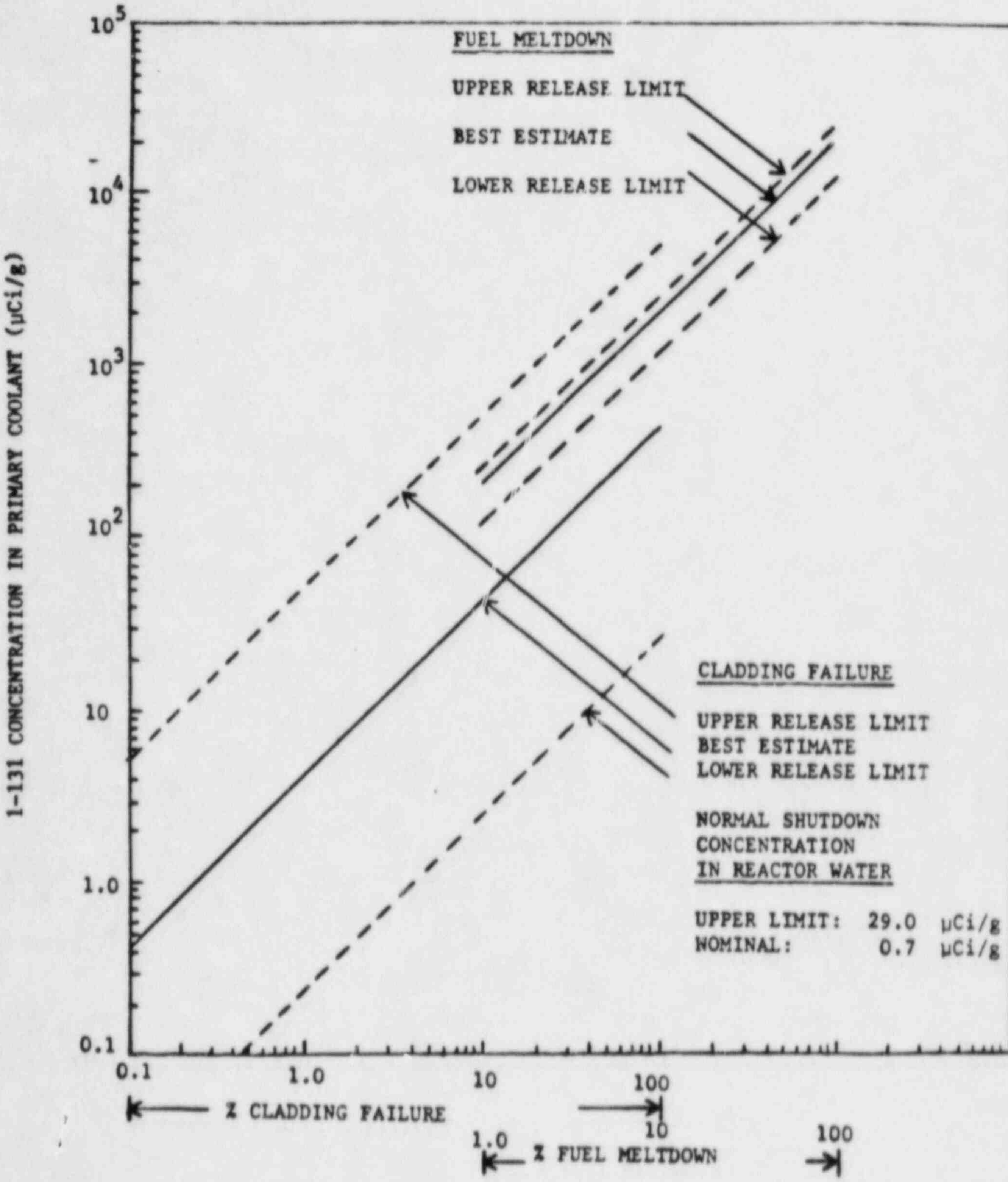
$$= \frac{\text{Iodine isotope concentration}}{\text{I-131 concentration}} \text{ for iodines}$$

PLANT PARAMETERS
(NEDO-2215)

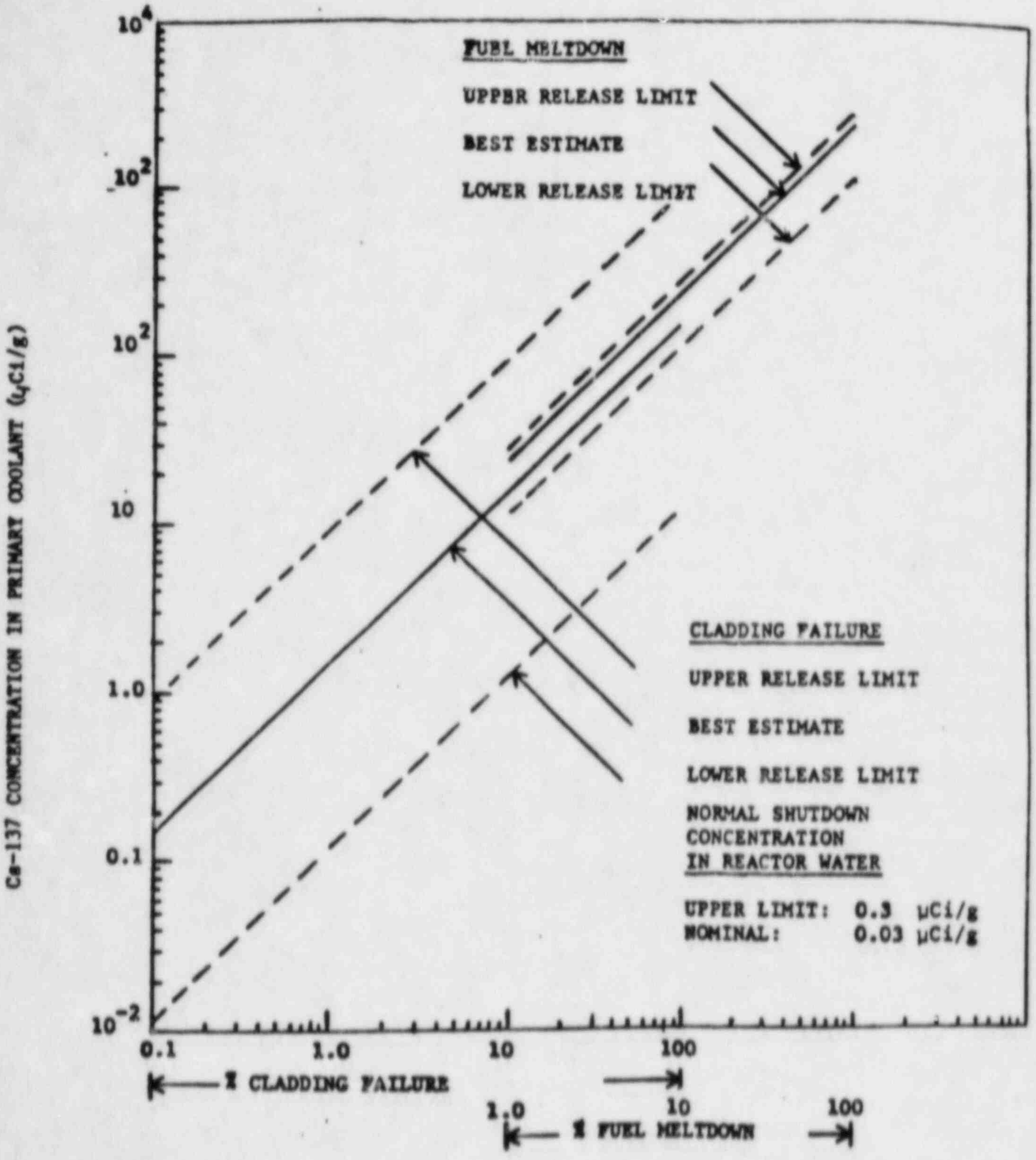
PLANT	REACTOR TYPE/CONTAINMENT DESIGN	RATED POWER (MWt)	PRIMARY COOLANT* REACTOR WATER MASS (10^8g)	PRIMARY COOLANT* SUPPRESSION POOL WATER (10^9g)	CONTAINMENT GAS* DRYWELL GAS VOL. (10^9cc)	CONTAINMENT GAS* TORUS/CONTAINMENT GAS VOLUME (10^9cc)
EF2	BWR 4 MKI	3292	2.77	3.23	4.64	3.71

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

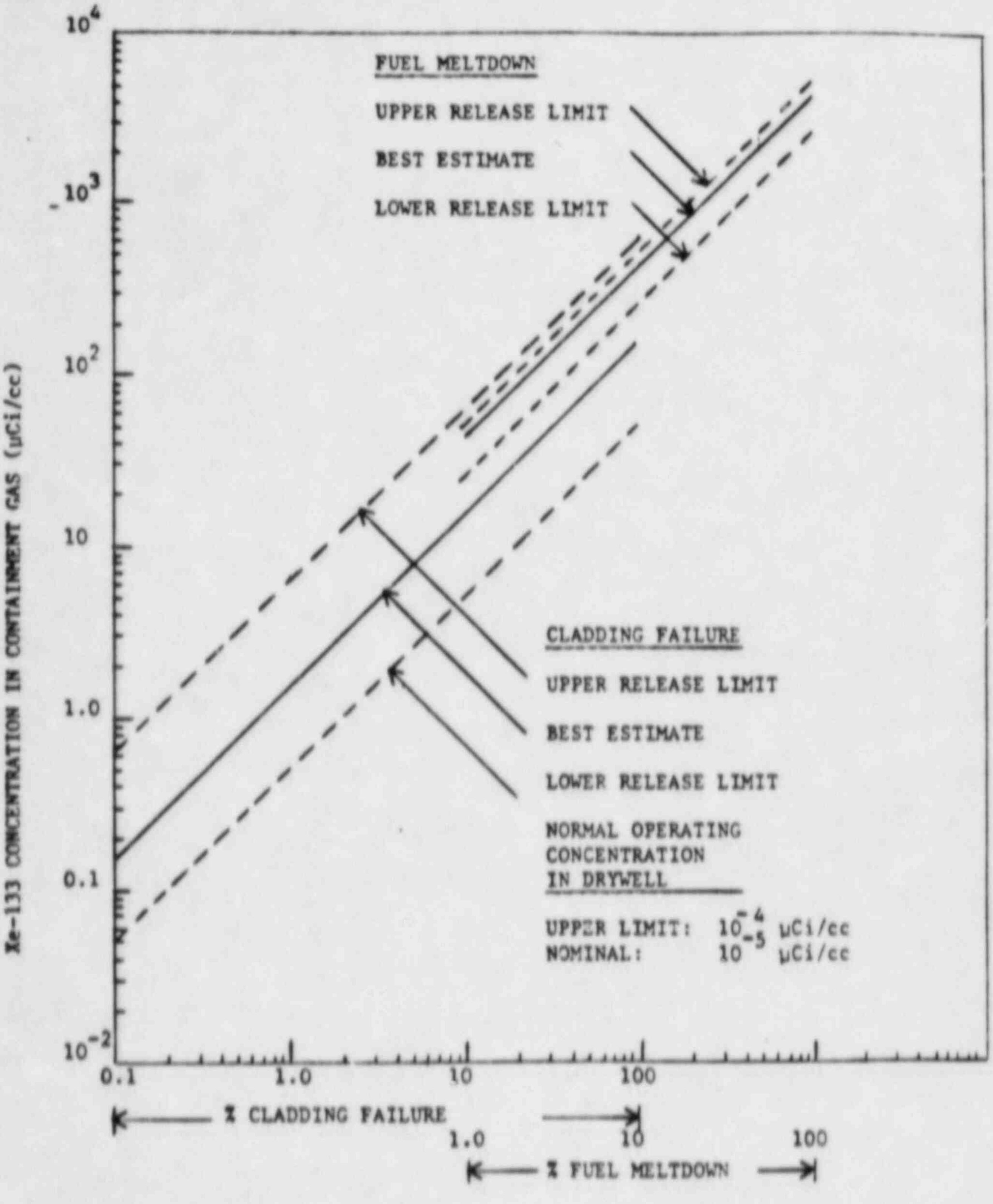
Total Containment Gas Volume - Drywell Gas + Torus (or Primary Containment in MKIII) gas



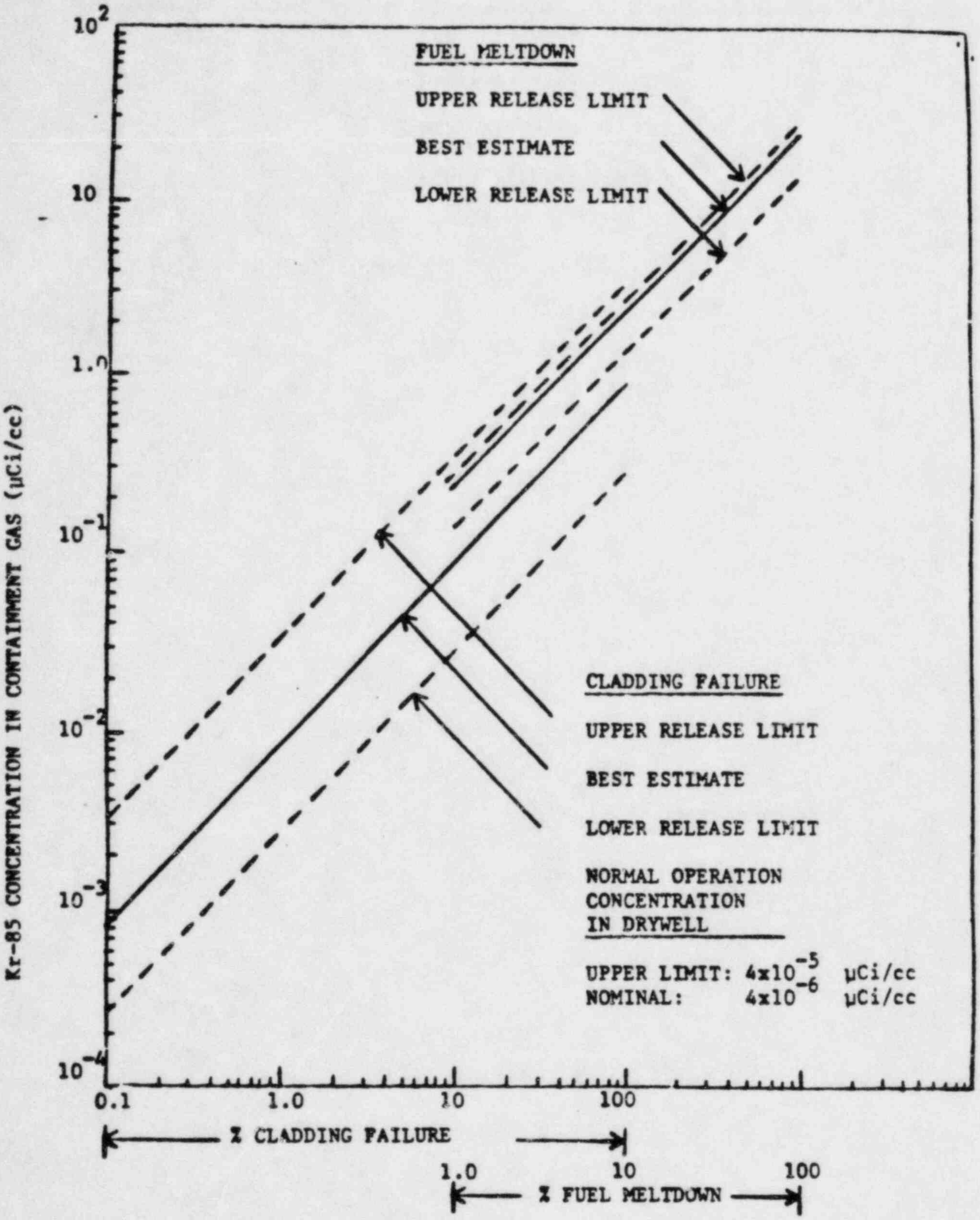
Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant



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



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

BEST-ESTIMATE FISSION PRODUCT RELEASE FRACTIONS

	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,HR)	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	0.380	0.855	---	---	---	0.190	0.190	0.190
Tellurium Group (Te,Se,Sb)	0.0001	$3x10^{-7}$	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)	$1x10^{-6}$	$3x10^{-9}$	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	---	---	---	---	---	---

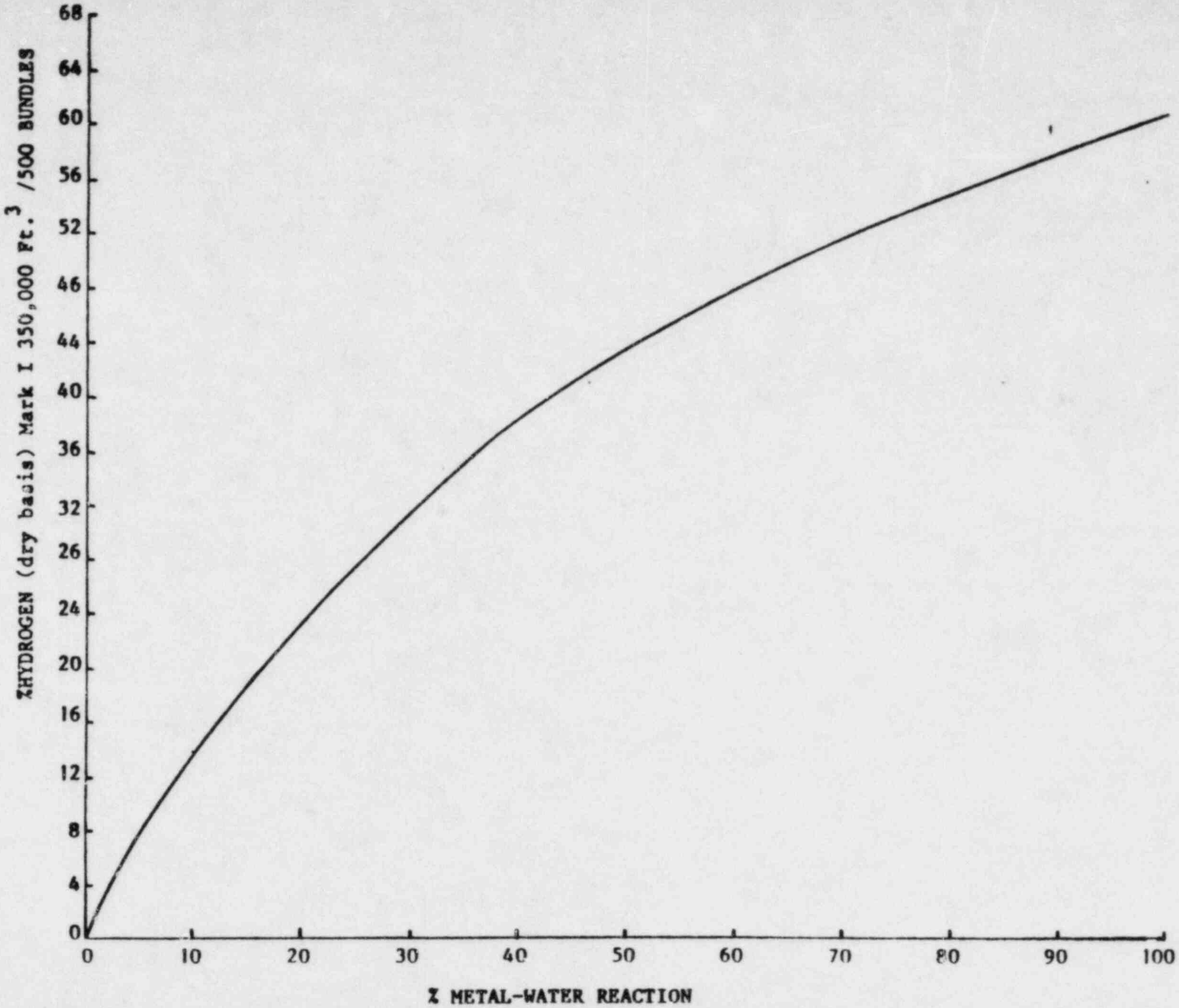
SAMPLES MOST REPRESENTATIVE OF CORE CONDITIONS DURING AN ACCIDENT FOR THE ESTIMATION OF CORE DAMAGE

Break Category/System Conditions	Sample Location: Jet Pump	Supp. Pool Liquid	Supp. Pool Atmos.	RHR	Drywell	Other Instructions
Small Liquid Line Break, Reactor Power ≥1%	Yes	---	Yes[1]	---	Yes[2]	
Small Liquid Line Break, Reactor Power <1%	---	---	Yes[1]	Yes	Yes[2]	A. RHR must be in shutdown cooling mode. B. Reactor water level must be raised and flow from moisture separators.
Small Steam Line Break, Reactor Power ≥1%	Yes	---	Yes[1]	---	Yes[2]	
Small Steam Line Break, Reactor Power <1%	---	---	Yes[1]	Yes	Yes[2]	A. RHR must be in shutdown cooling mode. B. Reactor water level must be raised and flow from moisture separators.
Large Liquid Line Break, Reactor Power ≥1%	Yes[3]	Yes[4]	Yes[1]	---	Yes[2]	A. Suppression pool must be in suppression cooling mode.
Large Liquid Line Break, Reactor Power <1%	---	Yes[4]	Yes[1]	Yes[3]	Yes[2]	A. RHR must be in shutdown cooling mode. B. Suppression pool must be in suppression cooling mode. C. Reactor water level must be raised and flow from moisture separators.

SAMPLES MOST REPRESENTATIVE OF CORE CONDITIONS DURING AN ACCIDENT FOR THE ESTIMATION OF CORE DAMAGE

Break Category/System Conditions	Sample Location: Jet Pump	Supp. Pool Liquid	Supp. Pool Atmos.	RHR	Drywell	Other Instructions
Large Steam Line Break, Reactor Power ≥1%	Yes[3]	Yes[4]	---	---	Yes	
Large Steam Line Break, Reactor Power <1%	---	---	Yes[1]	Yes	Yes[2]	A. RHR must be in shutdown cooling mode. B. Reactor water level must be raised and flow from moisture separators.

1. Use if SRV's are vented to the suppression pool.
2. Use if SRV's are not vented to suppression pool.
3. Use if makeup water flow is ≤50% of core flow present.
4. Use if makeup water flow is ≥50% of core flow present.



Hydrogen Conccentration for a Mark I Containment as a Function of Metal-Water Reaction

Note A-Analytical Assumptions

1. Containment Volume = 350,000 ft^3 (MK I-II)

2. Number of bundles = 500 (MK I-II)

3. Fuel type = 8 x 8 R

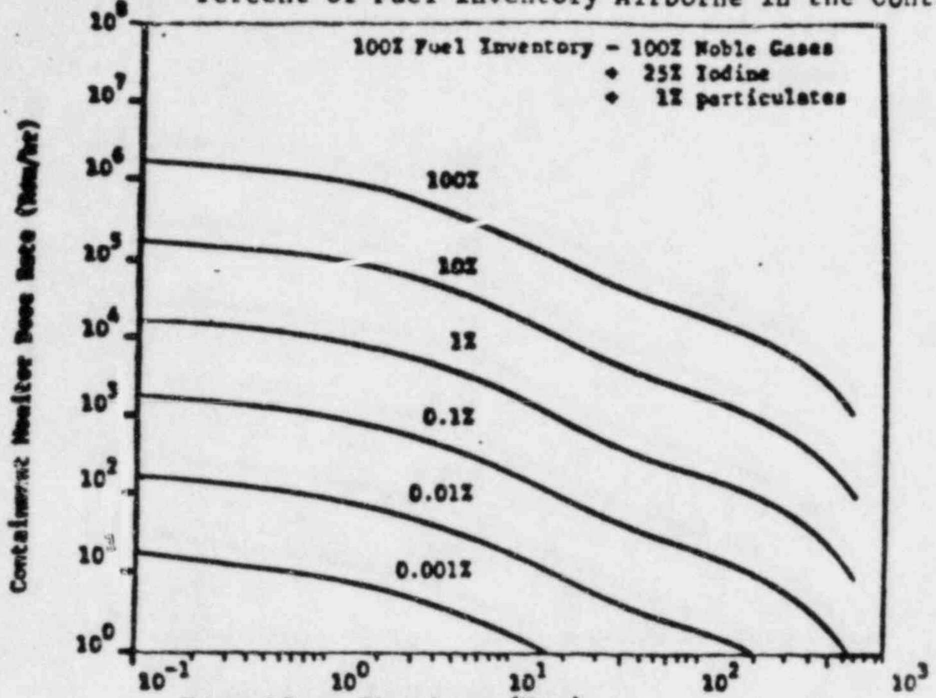
4. All hydrogen from metal-water reaction released to containment

5. Perfect mixing in containment

6. No depletion of hydrogen (e.g., containment leakage)

7. Ideal gas behavior in containment

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.

SEQUENCE OF ANALYSIS FOR ESTIMATION OF CORE DAMAGE