ENRICO FERMI ATOMIC POWER PLANT
UNIT NO. 2

INFORMATION ONLY

Type: CHEMISTRY PROCEDURES - SPECIAL TESTS

Title: DETERMINATION OF EXTENT OF CORE DAMAGE

RECORD OF APPROVAL AND CHANGES

Prepared by Robert Eberhardt/W. McNeil — April 23, 1982 (Date)

Submitted by [signature] (Section Head) — 4/29/82 (Date)

Recommended by [signature] (OSRO Chairman) — 4/29/82 (Date)

Recommended by E H Newton (Supervisor, Operational Assurance) — 4-29-82 (Date)

Approved by [signature] (Superintendent - Nuclear Production) — 4/29/82 (Date)

Revision No.	OSRO Recommended	Date	Supervisor, Operational Assurance Recommended	Date	Supt. - Nuc. Prod. Approved	Date
1						
2						
3						
4						
5						
6						
7						
8						

TABLE OF CONTENTS

TABLE OF CONTENTS

1.0 Purpose

The purpose of this procedure is to determine the degree of reactor core damage from the measured fission product concentrations in either 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 these 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 products in the core will be released when the fuel is melted at higher temperatures.

2.0 Discussion

The 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 3 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. 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 27hr 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 estimation of core damage.

3.0 References

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

3.2 Chemistry Procedure No. 78.000.14, Post Accident Sampling

3.3 Chemistry Procedure No. 70.000.05, Operation of the Chemistry ND6685

3.4 Chemistry Procedure No. 76.000.06, Operation of the Chemistry ND680

4.0 Equipment Required

4.1 Apparatus

4.1.1 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.

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

7.1.3 Correct the measured fission products for decay to the time of reactor shutdown.

7.1.4 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 Section 7.3.

7.1.5 Calculate the fission product inventory correction factor F_{Ii} per Section 7.4.

7.1.6 Calculate the plant parameter correction factors (Fg or Fw) per Section 7.5.

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

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

1. If the normalized concentrations, C_{wi}^{Ref} or C_{gi}^{Ref}, obtained in Section 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 27hr 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-75m	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 Section 7.7.

7.2.4 Determine release source by comparing results obtained in Section 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 temperature

7.4 Fission Product Inventory Correction Factor

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

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

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

T_j = duration of operating period j (days)*

T_j^0 = time between the end of operating period j time of reactor shutdown (days)*

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.

7.5 Plant Parameter Correction Factors

$$F_w = \frac{\text{EF2 coolant mass (g)}}{\text{reference plant coolant mass } (3.92\times10^9 \text{ g})}$$

$$F_g = \frac{\text{EF2 gas volume (cc)}}{\text{reference plant containment gas vol. } (4\times10^{10} \text{ cc})}$$

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:

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

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

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

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

λ_i = 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

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

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

8.0 Acceptance Criteria

None

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	---	---	4×10^{-5}*	4×10^{-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*

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

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

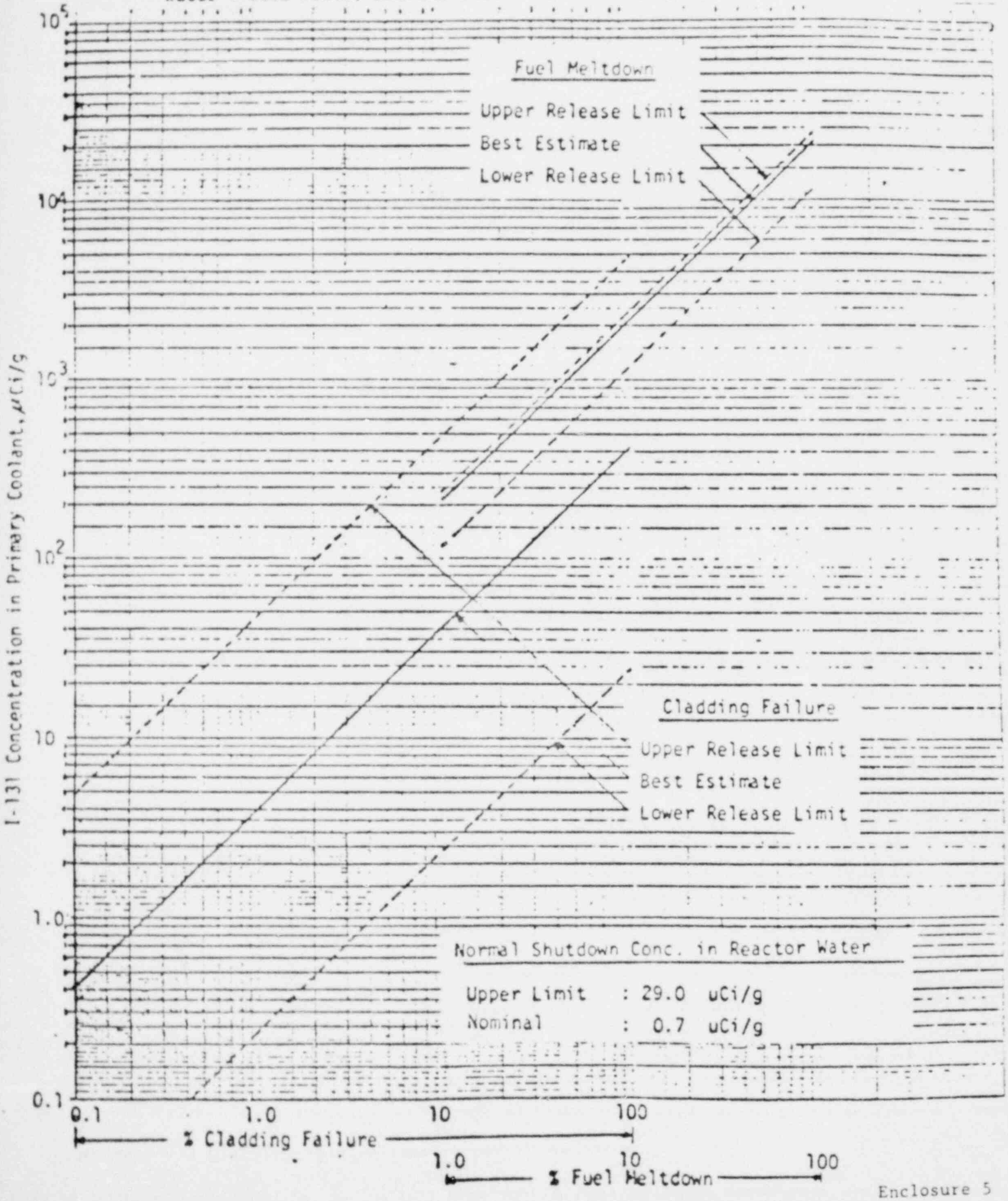
PLANT PARAMETERS

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

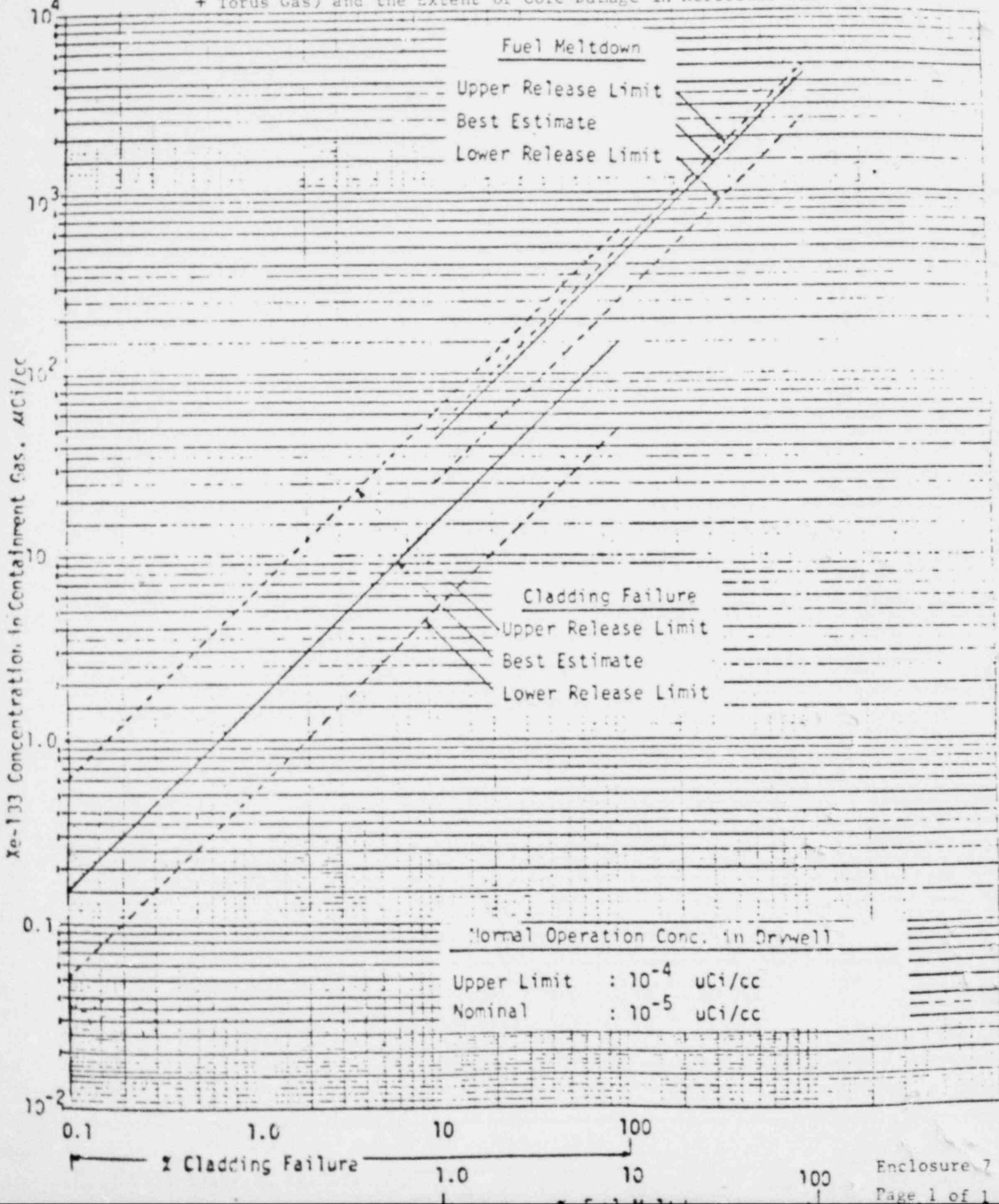
*Total Primary Coolant Mass = Reactor Water + Suppression Pool Water

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

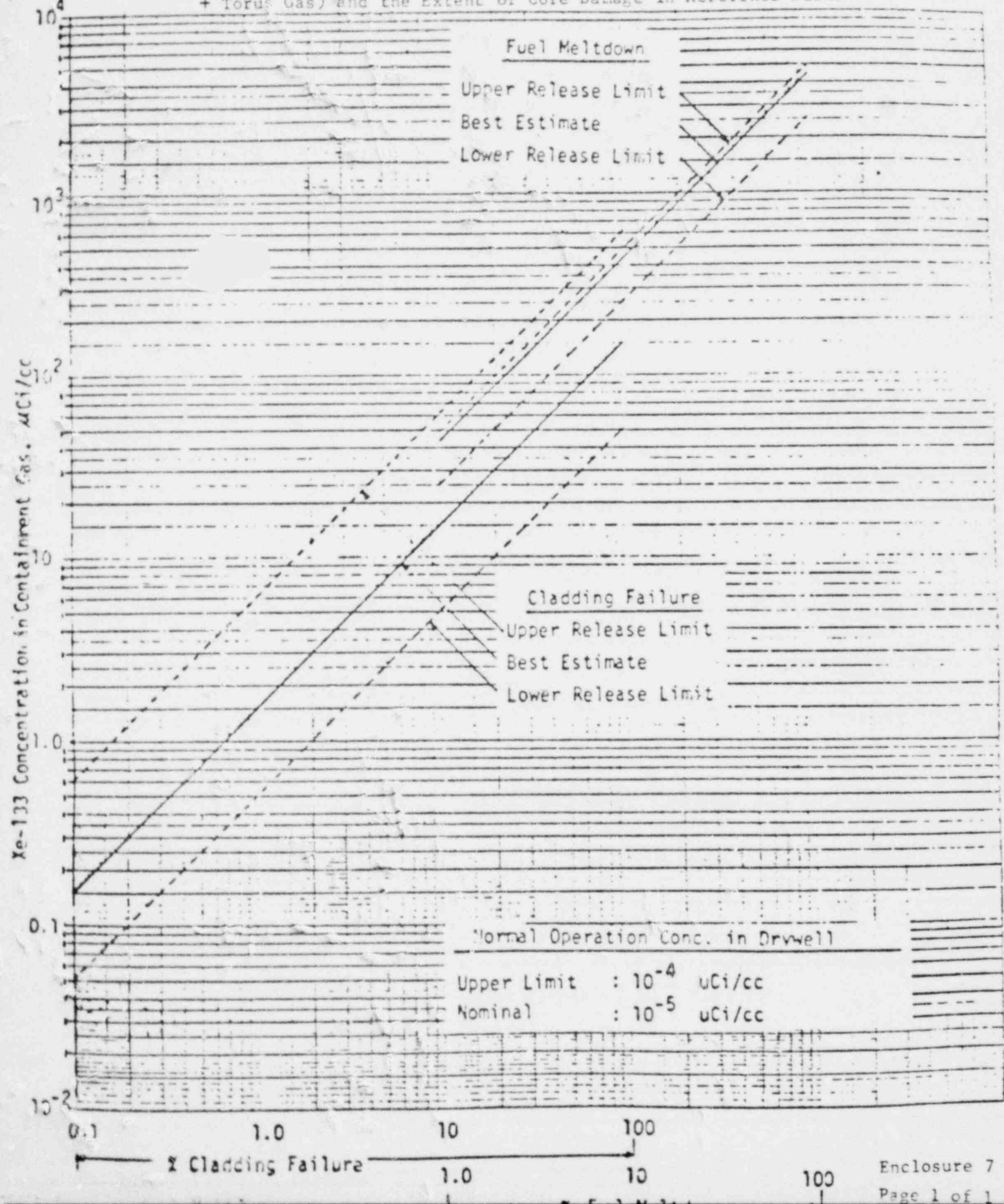
Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant



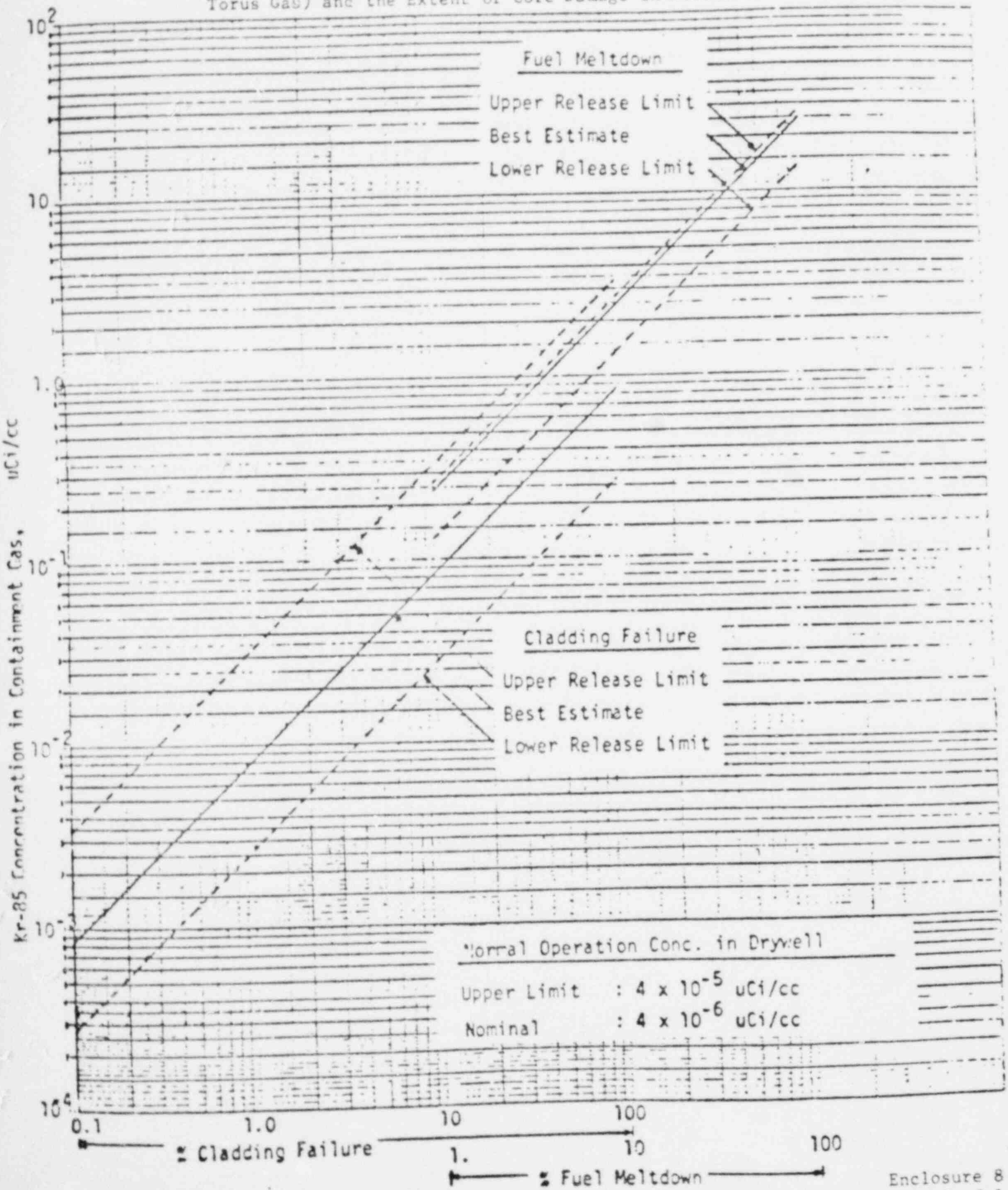
Relationship Between Xe-133 Concentration in the Containment Gas (Drywell + Torus Gas) and the Extent of Core Damage in Reference Plant



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[a]			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	0.380	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]	1. RHR must be in shutdown cooling mode. 2. 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]	1. RHR must be in shutdown cooling mode. 2. 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]	1. Suppression pool must be in suppression cooling mode.
Large Liquid Line Break, Reactor Power <1%	---	Yes[4]	Yes[1]	Yes[3]	Yes[2]	1. RHR must be in shutdown cooling mode. 2. Suppression pool must be in suppression cooling mode. 3. 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					Other Instructions
	Jet Pump	Supp. Pool Liquid	Supp. Pool Atmos.	RHR	Drywell	
Large Steam Line Break, Reactor Power $\geq$1%	Yes[3]	Yes[4]	---	---	Yes	
Large Steam Line Break, Reactor Power <1%	---	---	Yes[1]	Yes	Yes[2]	1. RHR must be in shutdown cooling mode. 2. 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 $\leq$50% of core flow present.
4. Use if makeup water flow is $\geq$50% of core flow present.