WNP-2 CORE DAMAGE EVALUATION

1.0 PURPOSE

The purpose of this procedure is to provide a method of estimating damage to the core following a reactor accident.

2.0 DISCUSSION

Core damage evaluation is based on several plant parameters including reactor water level, containment atmosphere radiation levels, containment hydrogen concentrations and fission product radionuclide concentrations in the reactor coolant and the containment atmosphere. The estimate of core damage is determined by comparing the levels and concentrations against those which could result from 100% core degradation at WNP-2.

The fission product inventory is based on 1095 days of operation at 3323 MWt or 100% rated power as calculated by General Electric[1] for concentrations of radionuclides in the coolant or the containment atmosphere. The radiation levels in the containment atmosphere are based on the same core power level but it is assumed to have operated only 80% of the time or 295 days per year for three years. These dose rates were calculated by ISOSHLD[2] a point kernel shielding code utilizing RIBD to develop the source terms. The two codes produce nearly identical source terms because of the relatively short after decay periods being considered. Other plant parameters are:

- o Number of fuel bundles = 764 bundles
- o Reactor Primary Coolant mass = 2.74×10^8g
- o Total Coolant (Primary plus suppression pool) = 3.17×10^9g
- o Containment Atmosphere (free volume) = 9.83×10^9cc
- o Drywell Atmosphere = 5.75×10^9cc

[illegible]

The procedure looks at water level history. If the core has been uncovered this would be a first indication that there may be fuel or fuel cladding damage. It looks at containment atmosphere radiation levels which will be quickest estimation of core damage in case of a LOCA. It also looks at the hydrogen present in the containment atmosphere. If hydrogen is present it will be indicative of the extent of metal/water reaction that has occurred with the zircalloy cladding surrounding the active fuel pellets. Gas and water samples may be taken of the reactor coolant and/or the containment atmosphere to be analyzed for fission product concentrations and/or hydrogen concentration. The presence of radioactive halogens and cesiums in the coolant and/or noble gases in the atmosphere will be indicative of core damage. The presence of less volatile fission products such as barium, lathenium or strontium would indicate fuel melting. Also, the ratio of short lived noble gas isotopes to Xe-133 and iodine isotopes to I-131 will assist in distinguishing between gas gap releases by cladding failure from core releases by fuel melt.

3.0 REFERENCES

1. H. A. Careway "Calculation of Fission Product Inventory and Spectra--RADC101 Program", NBDO-25176 (October 1980).

2. J. Greenborg et.al., "ISOSHLD - A Computer Code for General Purpose Isotope Shielding Analysis" BNWL-236 (June 1966).

3. C. C. Lin, "Procedures for the Determination of the Extent of Core Damage Under Accident Conditions" NEDO-22215 (August 1982).

4.0 PROCEDURE

4.1 Reactor Water Level

Determine the reactor water level history from the Fuel Zone Level monitor, recorder LR-R615 on panel P-601.

Full Scale	= -117"
Top of active core	= -167"
Bottom of active core	= -317"

From these readings determine how much, if any, of the core has been uncovered, for how long and when during the course of the accident.

4.2 Fission Gases in Containment

Take readings on the Containment Radiation LOCA monitors (located on panels RAD22 and RAD23). If radiation levels are high enough to be recognizable, utilizing Appendix A, determine the fraction of fission gases that have been released to containment.

4.3 Hydrogen Gas in Containment

Take readings from the containment Hydrogen Analyzers, located on panels K-I and K-II. Evaluate metal-water reaction in accordance with Appendix B to estimate the fraction of core cladding damaged. Top scale on the Analyzer is 10% H_2 which equates to 5% of the cladding.

If Analyzer is off-scale, hydrogen concentration will have to be determined from a containment atmosphere sample analyzed in the chem lab. Use Appendix B to evaluate extent of cladding damage.

4.4 Post Accident Sample

Under accident conditions when the WNP-2 Health Physics/Chemistry manager or designee has decided to take a sample:

4.4.1 Obtain reactor coolant and/or containment atmosphere samples, consistent with Appendix C, from the Post-Accident Sampling System.

4.4.2 Analyze the samples by gamma spectroscopy to determine the concentrations of radionuclides in each sample. Also analyze the atmosphere sample for hydrogen if required.

4.4.3 Normalize the concentration to a full power core ($3323 MW_t$) irradiated for 1095 days at decay time zero and for gas samples also to normalize to containment temperature and pressure in accordance with Appendix D correction factors.

4.4.4 Using normalized concentrations of:

Kr-85	Xe-133	I-131	Cs-134
Kr-88	Xe-135	I-133	Cs-137

enter the figures in Appendix E to determine the percent cladding failures or percent fuel melt.

4.4.5 If after entering figures in Appendix E the results could indicate either a gross cladding failure or a partial fuel melt, determine the amount of low volatile fission products present in the coolant sample. The absence of low volatile fission products would indicate gross cladding failures rather than melting.

4.4.6 To distinguish between cladding failures and melting, determine the ratio of Kr-87, Kr-88, Kr-85m to Xe-133 and/or I-132, 133, 134, 135 to I-131 and compare to ratios given in Appendix F.

APPENDIX A

CONTAINMENT RADIATION LEVELS

Four radiation monitors monitor the radiation levels inside the Drywell. Two of the detectors are located in the bioshield wall at elevations 522' and 525' and azimuth 60° (Figure 1) and 297° (Figure 2). They are in the best location to monitor the released fission gases in the drywell for the first two days following a reactor accident while radioactive gas energies are high. They also minimize the background from other sources such as plate-out and recirculation lines.

Two other detectors are located inside containment at elevation 515" and azimuth 51.5° (Figure 3) and 290° (Figure 4). These monitors will measure the long term and low energy radiation created by Xe-133 (80 KeV) and radiation from the solid radionuclides plated out on the interior surfaces.

The dose rate curves in the figures are based on the release of:

N.G. = 100% noble gas core (3323MW) inventory.

N.G.+I = 100% noble gas plus 50% halogen core (3323 MW) inventory.

To estimate the release of noble gases into the Drywell:

(1) Read the dose rate (D_1) for radiation monitors CONTAINMENT LOCA RAD-AZ-60° and CONTAINMENT LOCA RAD-AZ-297°, located on RAD-Boards 22 and 23 in the Control Room.

(2) On corresponding figures 1 and 2 enter the decay time since plant shutdown on the abscissa and determine meter reading (D_m) for the N.G. curve. If power level differed from 3323 MW, then correct D_m. Corrected $D_m = \frac{D_m}{3323}$ x (actual power level).

(3) Fraction (F_D) of noble gas core inventory released to the Drywell is:

$$F_D = \frac{D_1}{D_m} .$$

Example: o At 19 minutes decay, the reading on LOCA RAD AZ 60^o is 35.R/hr. Past operating level = 2700 MW.

o Enter figure #1 at 19 mins., to find a meter reading (D_m') of 2200R/hr.

o Corrected $D_m = \frac{2200}{3323} \times 2700 = 1788$

o $F_d = \frac{D_1}{D_m} = \frac{35}{1788} = 0.020$ or 2.0% core inventory.

(4) If the dose rate D_1 exceeds the N.G. curve then other radionuclides are present in significant quantity and core damage is more severe than just releasing the volatile noble gases.

The same method should be followed for the inside containment monitors using graphs on figures #3 and #4. These monitors should be utilized after 3 days decay.

FIGURE- 2

CONTAINMENT LOCA RAD AZ-297°

OUTSIDE CONTAINMENT MONITOR

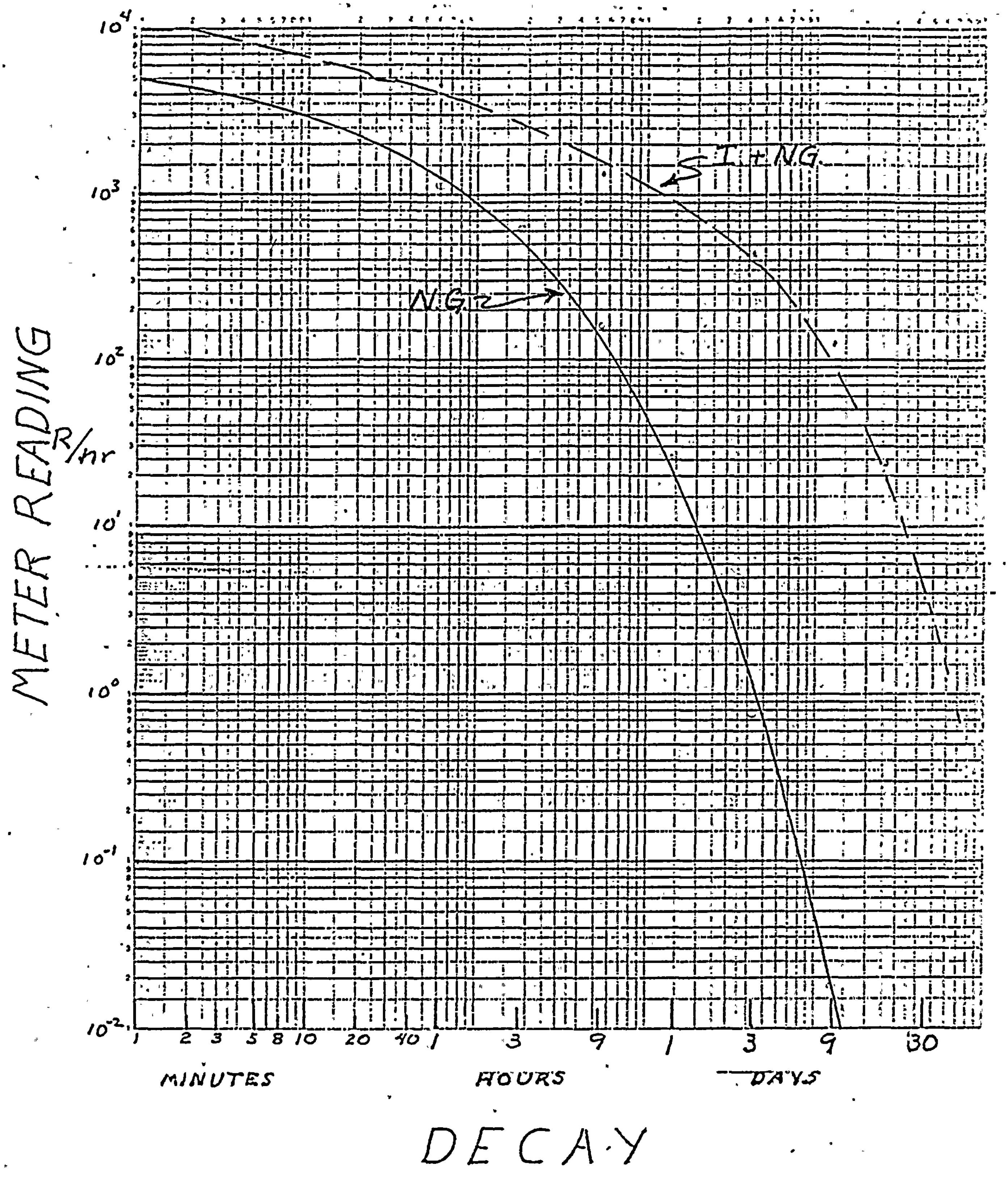


FIGURE I

CONTAINMENT LOCA RAD AZ - 60°

OUTSIDE CONTAINMENT MONITOR

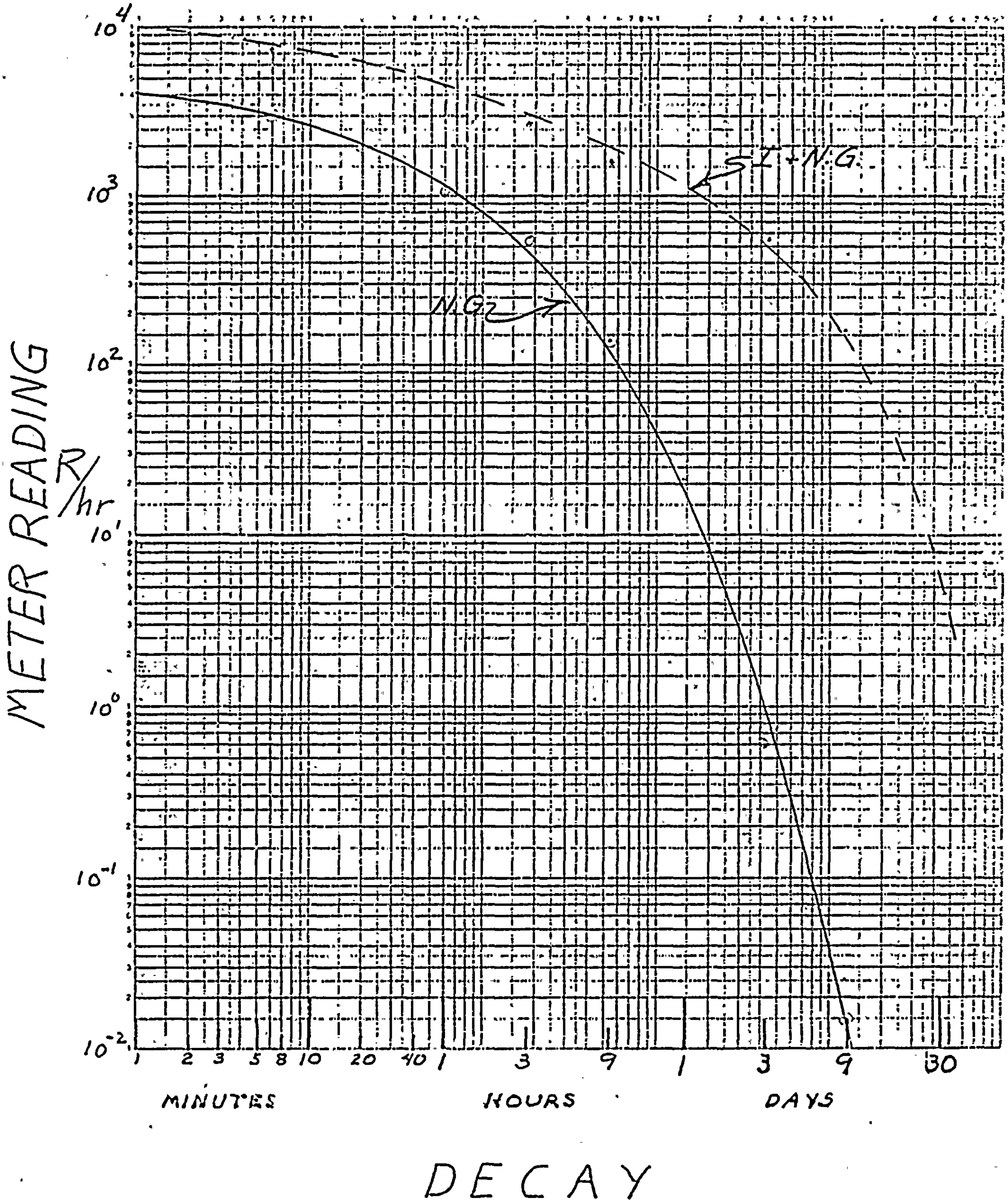
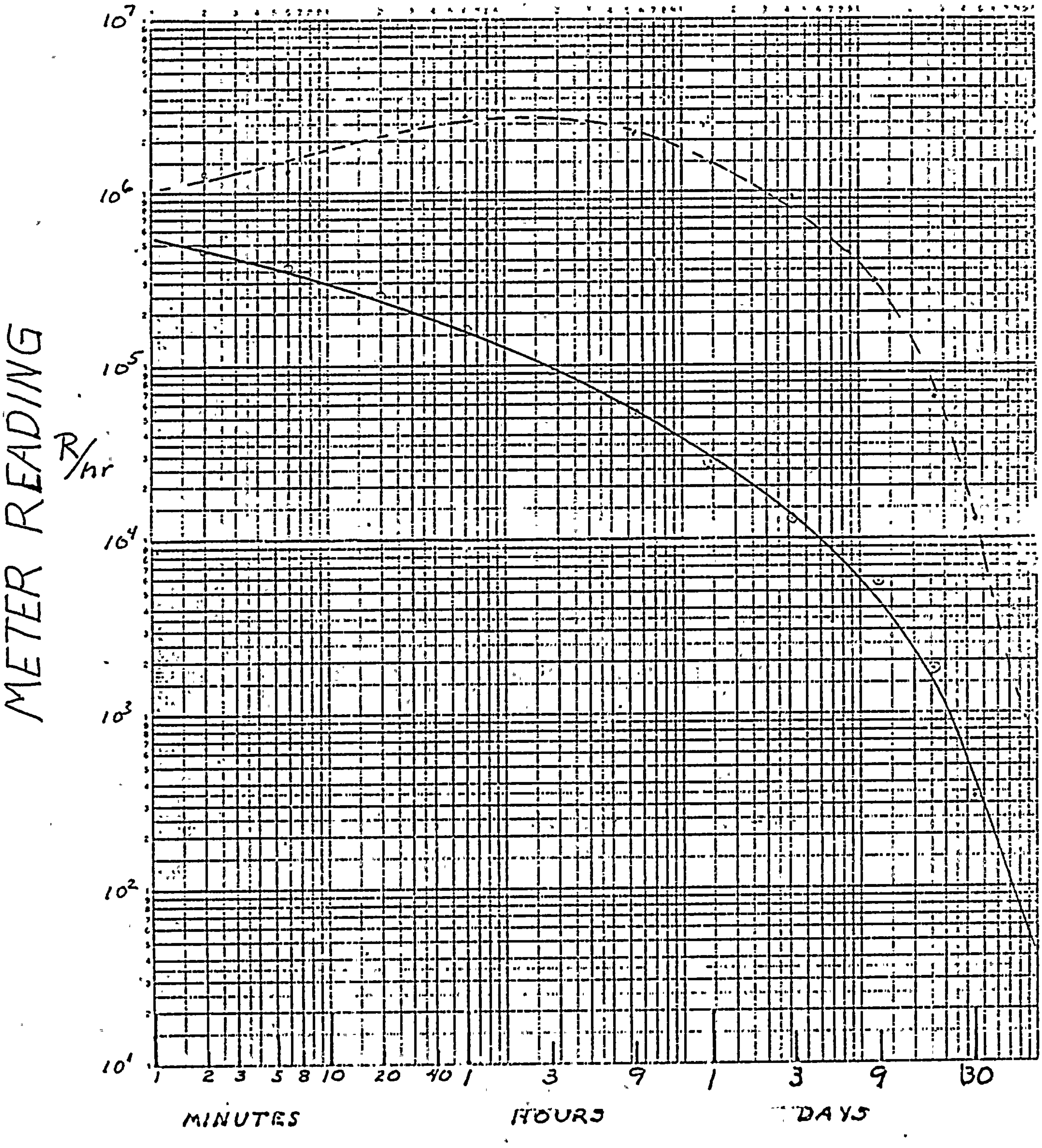


FIGURE 3

INSIDE CONTAINMENT
DETECTOR #3

AZ · 290°



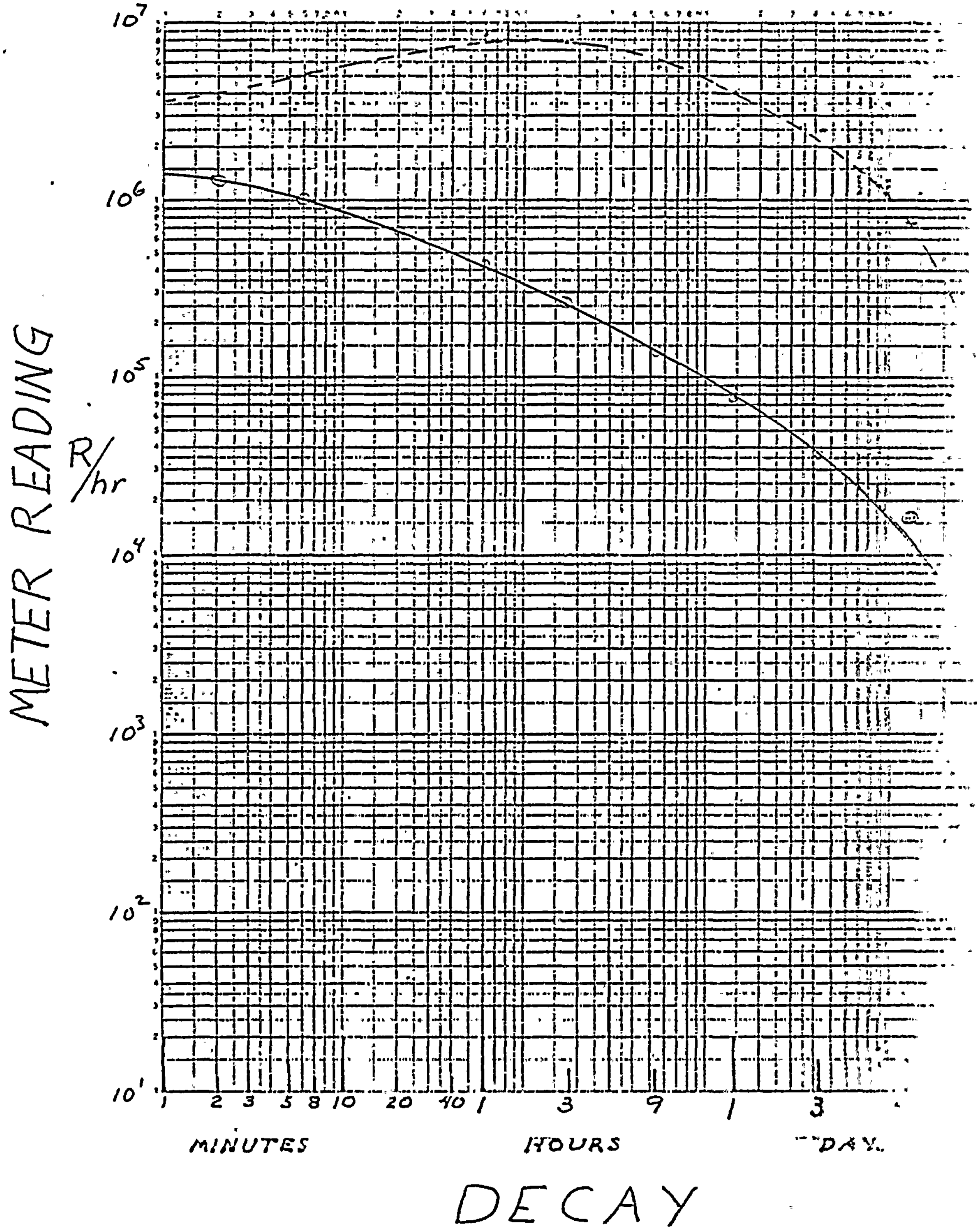
DECAY

FIGURE 4

INSIDE CONTAINMENT

DETECTOR #4

AZ 51.5°



APPENDIX B

CONTAINMENT HYDROGEN LEVELS

The extent of metal-water reaction for the zirconium fuel cladding can be estimated from the hydrogen content in the containment atmosphere. Containment hydrogen concentration can be determined from the containment hydrogen-oxygen analyzers up to 10%. Analysis of concentrations greater than 10% H_2 requires analysis of samples taken by the post accident sampler.

To determine the extent of zirconium metal-water reaction:

(1) Enter Figure B-1 with the percent hydrogen and from the curve determine the percent metal water reaction.

ANALYTICAL ASSUMPTIONS

(1) Containment volume = Drywell - 200,450 ft^3 + Wetwell - 142,500 ft^3
= Containment total - 343,000 ft^3

(2) Active zirconium--only that portion surrounding fuel pellets

- Length = 150"
- I.D. = 0.419"
- Wall = 0.032"
- Density = 0.236 lb/in^3
- 62 tubes/fuel assembly
- 764 assemblies

(3) $Zr + 2\ H_2O \quad ZrO_2 + 2H_2\uparrow$

(4) 100% metal-water reaction produces 7.56×10^6 moles H_2

(5) Amount of N_2 in containment when isolated assuming that the Drywell is at $135^{\circ}F$ and 30% humidity and the Wetwell is at $90^{\circ}F$ and 100% humidity.

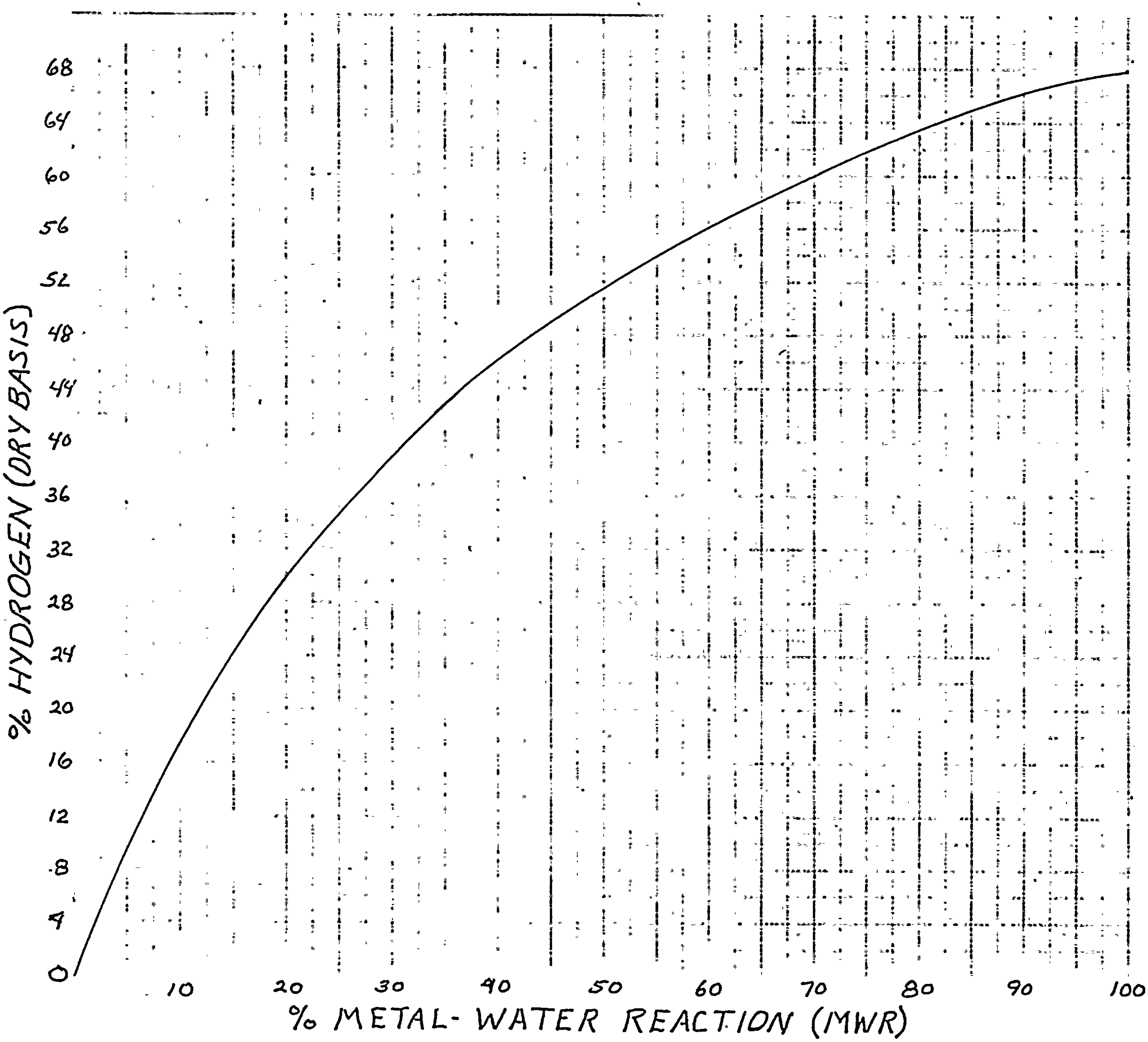
$$\text{Net } N_2 = 3.53 \times 10^5 \text{ moles}$$

(6) Analysis of H_2 is based on a dry sample.

(7) All hydrogen from metal water reaction is released into containment.

(8) It is mixed uniformly throughout drywell and wetwell.

FIGURE B-1



% HYDROGEN (DRY BASIS)
68
64
60
56
52
48
44
40
36
32
28
24
20
16
12
8
4
0
10
20
30
40
50
60
70
80
90
100
% METAL-WATER REACTION (MWR)

APPENDIX C

OBTAINING A REPRESENTATIVE SAMPLE
FOR ESTIMATING THE CORE'S CONDITION POST-ACCIDENT

(1) Coolant samples may be obtained from the:

- o Reactor vessel at jet pump #10 or #20 instrument tap
- o RHR downstream of Pump A or B
- o Suppression pool

(2) Atmosphere samples may be obtained from the:

- o Drywell above seal plate (hydrogen sampling)
- o Drywell below seal plate
- o Wetwell

Before withdrawing a sample consider the following (this is not an all inclusive list):

(1) Have Safety Relief Valves (SRVs) lifted?

(2) Temperature and pressure in Drywell and Wetwell.

(3) Is HPCS operating?

(4) Is LPCS operating?

(5) Are RHR pumps operating? Which mode?

- o Shutdown cooling (normal) mode.
- o Shutdown cooling (LPCI) mode.
- o Suppression cooling mode.
- o Drywell or wetwell spray mode.

(6) Drywell radiation levels.

(7) Reactor vessel water level (fuel zone level monitor).

(8) If recirculation pumps are not operating it is necessary to raise water level so the flow through the core overflows the moisture separators (+51" or 15" above normal water level) to be able to obtain a representative sample of the jet pump. (One exception is for a large liquid line break where there is a reverse flow through core.)

OBTAINING A REPRESENTATIVE SAMPLE
OF THE CORE COOLANT TO ESTIMATE CORE DAMAGE

Break Category	Sample Location to be Used					
*RCS pressure	Jet Pump	Supp. Pool	RHR Pump	Drywell ATMS	Wetwell ATMS	Other Instructions
No break or small break						
Hi press.	Yes	-	-	A	B	
Lo press.	Yes	-	Yes	A	B	o RHR in shutdown cooling mode
Large Break Liquid line						
Hi press.	C	D	D	A	B	o RHR in Suppression pool cooling
Lo press.	-	D	Yes	A	B	o RHR in shutdown cooling and/or suppression pool cooling modes.
Large Break Steam Line						
Hi press.	C	D	D	Yes	-	o RHR in suppression pool cooling mode
Lo press.	Yes	-	Yes	A	B	o RHR in shutdown cooling mode.

*Low pressure is when pressure is low enough for RHR shutdown cooling mode to be utilized.

Notes: A - Use if SRVs are not vented
B - Use if SRVs vent to suppression pool
C - Use if make-up water is 50% of core flow
D - Use if make-up water is 50% of core flow

APPENDIX D

This Appendix contains:

- o Reference Plant Parameters
- o Baseline fission product (f.p.) concentrations in Reactor Coolant and Drywell atmosphere under normal conditions
- o Core Fission Product Inventory
- o Normalization of Measured Fission Product Concentrations

REFERENCE PLANT PARAMETERS
(WNP-2)

			Mass
Power Level	= 3323MW$_t$	Reactor Coolant	= 2.74×10^8g
		Coolant + Suppression Pool	= 3.44×10^9g
Fuel bundles	= 764		
			Volume
		Drywell Atmosphere	= 5.68×10^9cc
Core operating time	= 1095 days	Total containment	= 9.71×10^9cc

Table D-1
Base Line

FISSION PRODUCT CONCENTRATIONS IN REACTOR COOLANT AND DRYWELL ATMOSPHERE DURING REACTOR SHUTDOWN UNDER NORMAL CONDITIONS

	Reactor Coolant, μCi/g		Drywell Atmosphere (μCi/cc)	
Isotope	Upper Limit	Nominal	Upper Limit	Nominal
I-131	29	0.7	---	---
Cs-137[c]	0.3[a]	0.03[b]	---	---
Xe-133	---	---	10^{-4}[a]	10^{-5}[b]
Kr-85	---	---	4×10^{-5}[a]	4×10^{-6}[b]

[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.

Table D-2

CORE INVENTORY OF MAJOR FISSION PRODUCTS IN
WNP-2 PLANT OPERATED AT 3323 MWt FOR THREE YEARS (1095 DAYS)

Chemical Group	Isotope*	Half-Life (Hours)	Inventory** 10^6 Ci	Major Gamma Ray Energy (Intensity) KeV (γ/d)
Noble Gases	Kr-85m	4.48	22.	151(0.753)
	Kr-85	9.39×10^4	1.0	514(0.0044)
	Kr-87	1.27	43.	403(0.495)
	Kr-88	2.84	61.	196(0.26), 1530(0.109)
	Xe-133	126.	184.	81(0.365)
	Xe-135	9.09	24.	250(0.899)
Halogens	I-131	193.	87	364(0.812)
	I-132	2.29	127.	668(0.99),773(0.762)
	I-133	20.8	183.	530(0.86)
	I-134	.877	201.	847(0.954),884(0.653)
	I-135	6.59	172.	1132(0.225),1260(0.286)
Alkali Metals	Cs-134	1.80×10^4	18.	605(0.98),796(0.85)
	Cs-137	2.64×10^5	11.	662(0.85)
	Cs-138	0.537	162.	463(0.307),1426(0.76)
Tellurium Group	Te-132	78.0	125.	228(0.88)
Noble Metals	Mo-99	66.0	167.	740(0.128)
	Ru-103	946.	141.	497(0.89)
Alkaline Earths	Sr-91	9.52	105.	750(0.23),1024(0.325)
	Sr-92	2.71	112.	1388(0.9)
	Ba-140	307.	157.	537(0.254)
Rare Earths	Y-92	1406.	107.	934(0.139)
	La-140	40.2	167.	487(0.455),1597(0.955)
	Ce-141	780.	147.	145(0.48)
	Ce-144	6830.	117	134(0.108)
Refractories	Zr-95	1104	147.	724(0.437),757(0.553)
	Zr-97	16.8	151.	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.

NORMALIZATION OF MEASURED FISSION PRODUCT CONCENTRATIONS

The measured isotope concentrations in the coolant and airborne activity as received from the chemical lab must be normalized to referenced plant conditions for use in Appendix E to determine core damage. The referenced plant core fission product inventory is WNP-2 operating at 3323 MW_t for 1095 days. The coolant mass and the contained air volume are given under Plant Parameters (page D-1). The following four correction factors as determined in steps (1), (2), (3) and (4) are applied to the measured individual isotope concentrations as shown in step (5) to give concentrations normalized to the reference plant.

(1) Correction for Core Operating History (F_I)

The inventory of fission product isotopes given in Table D-2 is for WNP-2 core operated at 3323MW for 1095 days at zero decay.

- For short-lived radionuclides if the last operating period equals more than 5 half-lives then correct for power

$$F_{IP} = \frac{3323}{P}$$

where P = operating power level.

- For longer lived radionuclides correct for actual length of operating period and the power level.

$$F_{IO} = \frac{1-e^{-\lambda 1095}}{1-e^{-\lambda T}} \times \frac{3223}{P}$$

Where $\lambda = \frac{0.693}{t_{1/2}}$ and $t_{1/2}$ = half life (days)

T = Operating period in days.

If the operating period exceeds 5 half lives this correction is not needed.

(E.g., if the operating period (T) exceeds 40 days no correction needed for I-131 ($t_{1/2}$ = 8.0 days) or any isotope with a shorter half life.)

o For very long-lived radionuclides (i.e., Cs-137, Kr-85, Sr-90)

This takes into account the full core history of operating periods, power levels and shutdowns (see example calc.).

$$F_{IS} = \frac{3323\ (1-e^{-\lambda 1095})}{\Sigma_J \left[P_J \left(1-e^{-\lambda T_J}\right)\left(e^{-\lambda T_J^o}\right)\right]}$$

J = summation of all operating periods

P_J = Power level during operating period J (MW_t)

T_J = Operating time for period J (days)

T_J^o = Time since end of operating period J

SAMPLE CALCULATION FOR OPERATING HISTORY

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

o For I-131 = $\frac{0.693}{8.04}$ = 0.0862 day^{-1}

$$F_{I(I-131)} = \frac{3323(1-e^{-0.0862x1095})}{1000(1-e^{-0.0862x60})e^{-0.0862x254} + 2000(1-e^{-0.0862x200})}$$

$$\overline{e^{-0.0862x44} + 3000(1-e^{-0.0862x14})e^{-0.0862x0}}$$

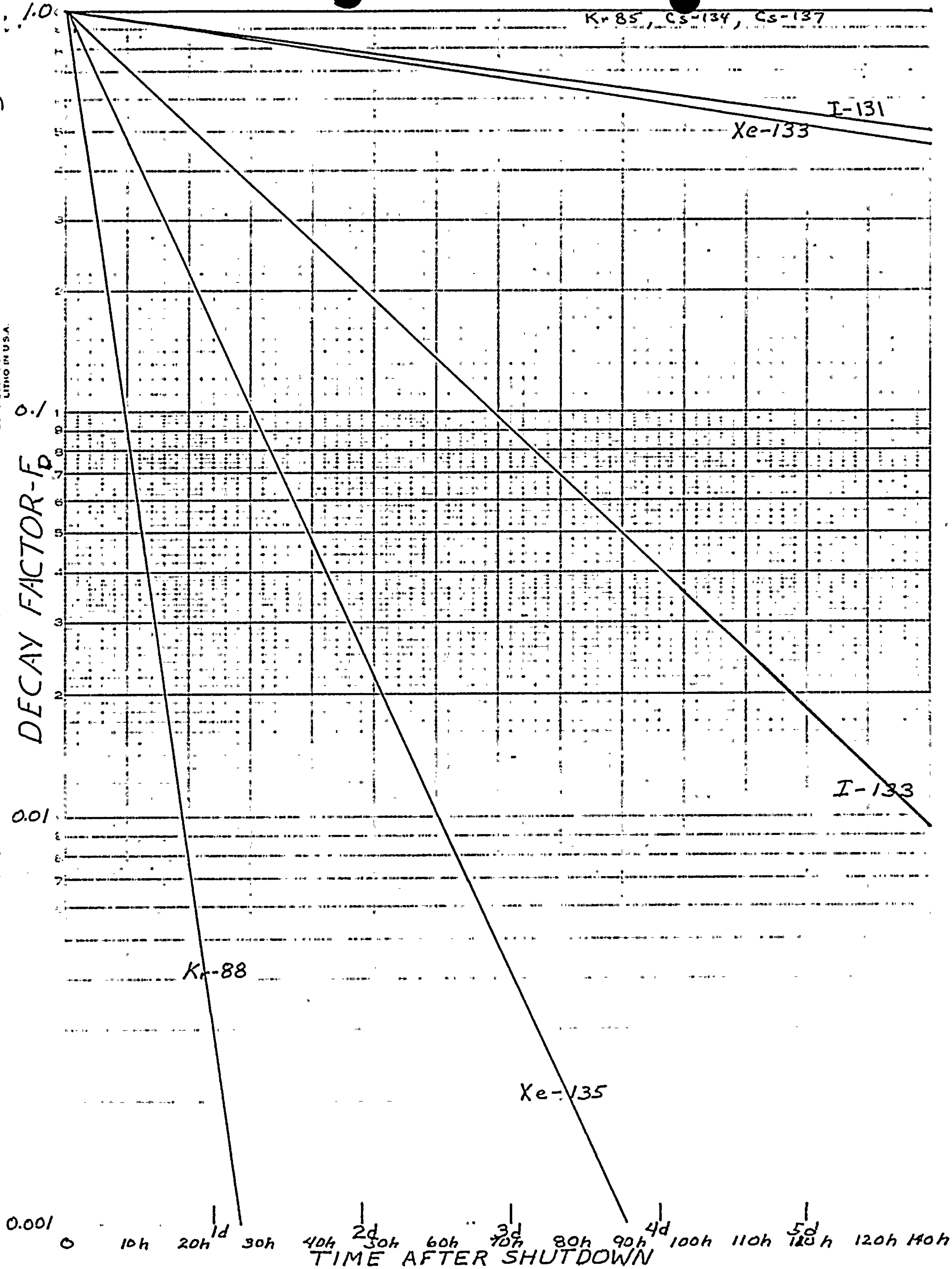
$$= \frac{3651}{0 + .45 + 2103} = 1.7$$

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

$$F_{I(Cs-137)} = \frac{3323(1-e^{-6.29\times10^{-5}\times1095})}{1000(1-e^{-6.29\times10^{-5}\times60})e^{-6.29\times10^{-5}\times254} + 2000(1-e^{-6.29\times10^{-5}\times200})e^{-6.29\times10^{-5}\times44} + 3000(1-e^{-6.29\times10^{-5}\times14})e^{-6.29\times10^{-5}\times0}}$$

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

FIGURE D-1



Kr 85, Cs-134, Cs-137
I-131
Xe-133
I-133
Kr-88
Xe-135
DECAY FACTOR-F_D
1.0
0.1
0.01
0.001
0
10h
20h
1d
30h
40h
2d
50h
60h
3d
70h
80h
90h
4d
100h
110h
5d
120h
130h
140h
TIME AFTER SHUTDOWN

(2) Correction for Radioactive Decay (F_D) (Also see Figure D-1)

$$F_D = \frac{C_o}{C}$$

$$C_o = Ce^{\lambda T} = C \times e^{\frac{0.693}{t_{1/2}} \times T}$$

C_o = Concentration, μCi/g or μCi/cc, at time of shutdown

C = Concentration at time measured (t) after shutdown

T = time sample concentration is measured (hours after shutdown).

$t_{1/2}$ = half life - hours

Radionuclide	t 1/2 Half-life Hours	λ (hr^{-1})
Kr-85	9.39 + 4*	7.38-6
Kr-88	2.84	0.244
Xe-133	126.	5.50-3
Xe-135	9.09	0.0763
I-131	193.	3.59-3
I-13	20.8	0.0333
Cs-134	1.80 + 4	3.85-5
Cs-137	2.64 + 5	2.63-6

*Note: 9.39+4 is same as 9.39×10^4

(3) Correction for Gas Sample Vial Temperature and Pressure, (F_{TP})

$$F_{TP} = \frac{P_c \times Tv}{P_v \times Tc} \qquad C_c = C_v F_{TP}$$

C_c, P_c, T_c = Concentration, Pressure and Temperature in in containment

C_v, P_v, T_v = Concentration, Pressure and Temperature in vial

(4) Measured Concentrations (c)

Measured concentrations reported from the Chemical Lab include:

C_c = reactor system coolant - $\mu Ci/g$

C_s = suppression pool water - $\mu Ci/g$

C_d = drywell air - $\mu Ci/cc$

C_{sp} = Suppression Pool Air - $\mu Ci/cc$

When both the reactor coolant concentration (C_c) and the suppression pool concentration (C_s) are measured, they are to be averaged to give a total water concentration (C_T).

$$C_T = \frac{(C_c \times 2.74 \times 10^8 g) + (C_s \times 3.17 \times 10^9 g)}{(3.44 \times 10^9 g)}$$

When both the drywell concentration (C_d) and the suppression pool free air concentration (C_{sp}) are measured, they are to be averaged to give a containment air concentration (C_{ct}).

$$C_{ct} = \frac{(C_d \times 5.75 \times 10^9 cc) + (C_{sp} \times 4.08 \times 10^9 cc)}{(9.83 \times 10^9 cc)}$$

(5) Normalized Concentration (C_{NW} - coolant, C_{NG} - gas)

To determine the normalized concentration, multiply the measured concentration by the appropriate correction factors shown below:

$C_{NW} = C_T \times F_I \times F_D$	Reactor coolant + suppression pool
$= C_c \times F_I \times F_D$	Reactor coolant
$= C_s \times F_I \times F_D$	Suppression pool

$C_{NG} = C_{ct} \times F_I \times F_D \times F_{TP}$ Containment (total)

$= C_d \times F_I \times F_D \times T_{TP}$ Drywell

$= C_{sp} \times F_I \times F_{TP}$ Suppression pool Free Air volume

APPENDIX E

The following figures are plots of the release fraction for individual isotopes from the gas gap or from a fuel melt condition for WNP-2 as if it had operated at 3323 MW_t for 1095 days. This serves as the normalized condition.

The solid line indicates the "best estimate"(3) with dash lines indicating the upper and the lower limit.

From Appendix D part 5, enter the airborne normalized concentration (C_{NW}) in Ci/cc or the coolant normalized concentration (C_{WG}) in uCi/g along the ordinate and determine the percent cladding failed or the percent fuel melt along the abscissa.

There are four sets of figures covering eight individual fission product radionuclides as follows:

Drywell (5.68×10^9cc)	Half Life	Containment (9.71×10^9cc)
Kr-85	10 yr	Kr-85
Kr-88	2.84 hr	Kr-88
Xe-133	5.25 day	Xe-133
Xe-135	9.11 hr	Xe-135

Reactor Coolant (2.74×10^8g)	Half Life	Reactor Coolant Plus Suppression Pool (3.44×10^9g)
I-131	8.04 day	I-131
I-133	20.8 hr	I-133
Cs-134	2.06 yr.	Cs-134
Cs-137	30 yr.	Cs-137

FIGURE E-1
Kr-85 (Drywell)

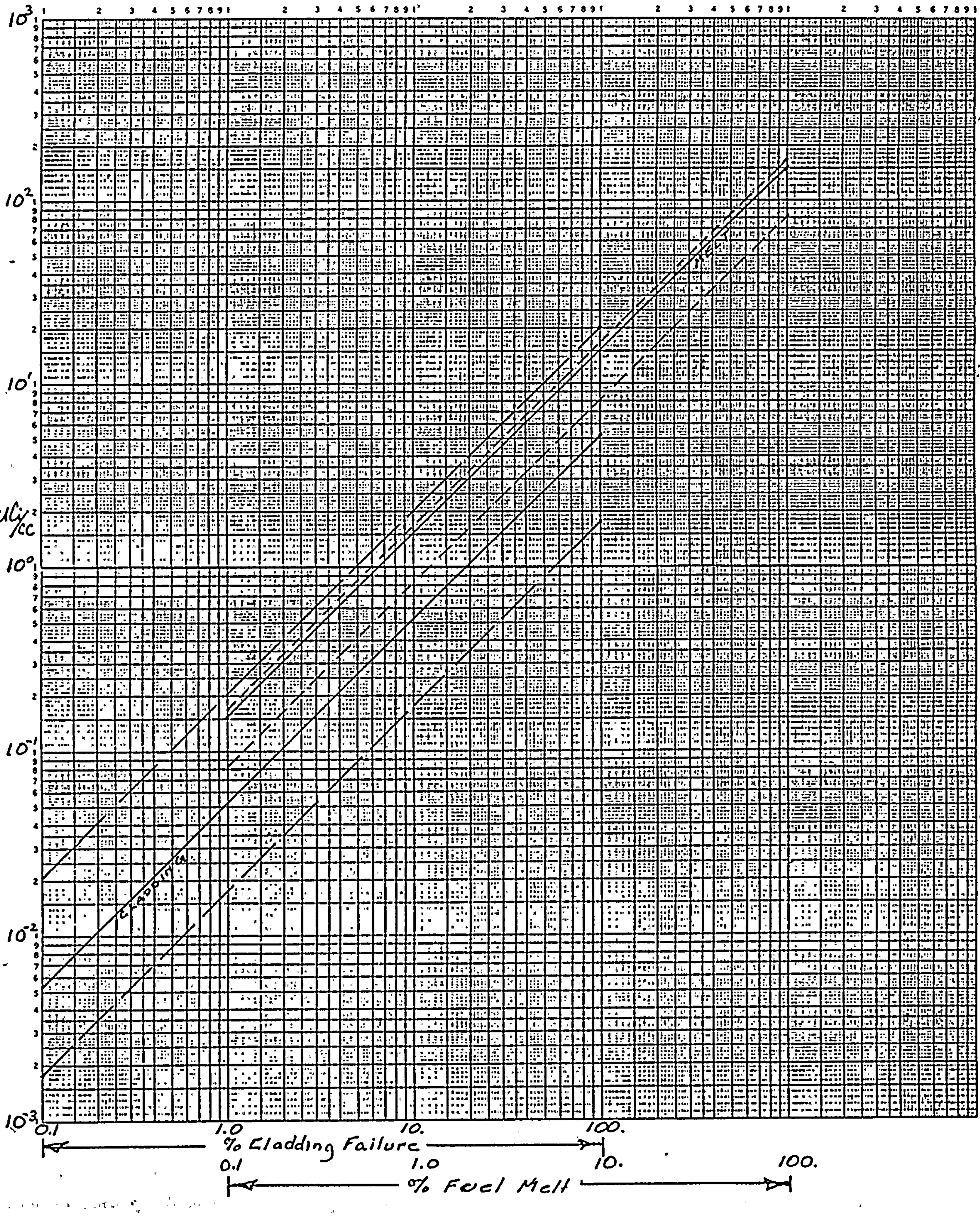


FIGURE E-2
Kr-88 (Drywell)

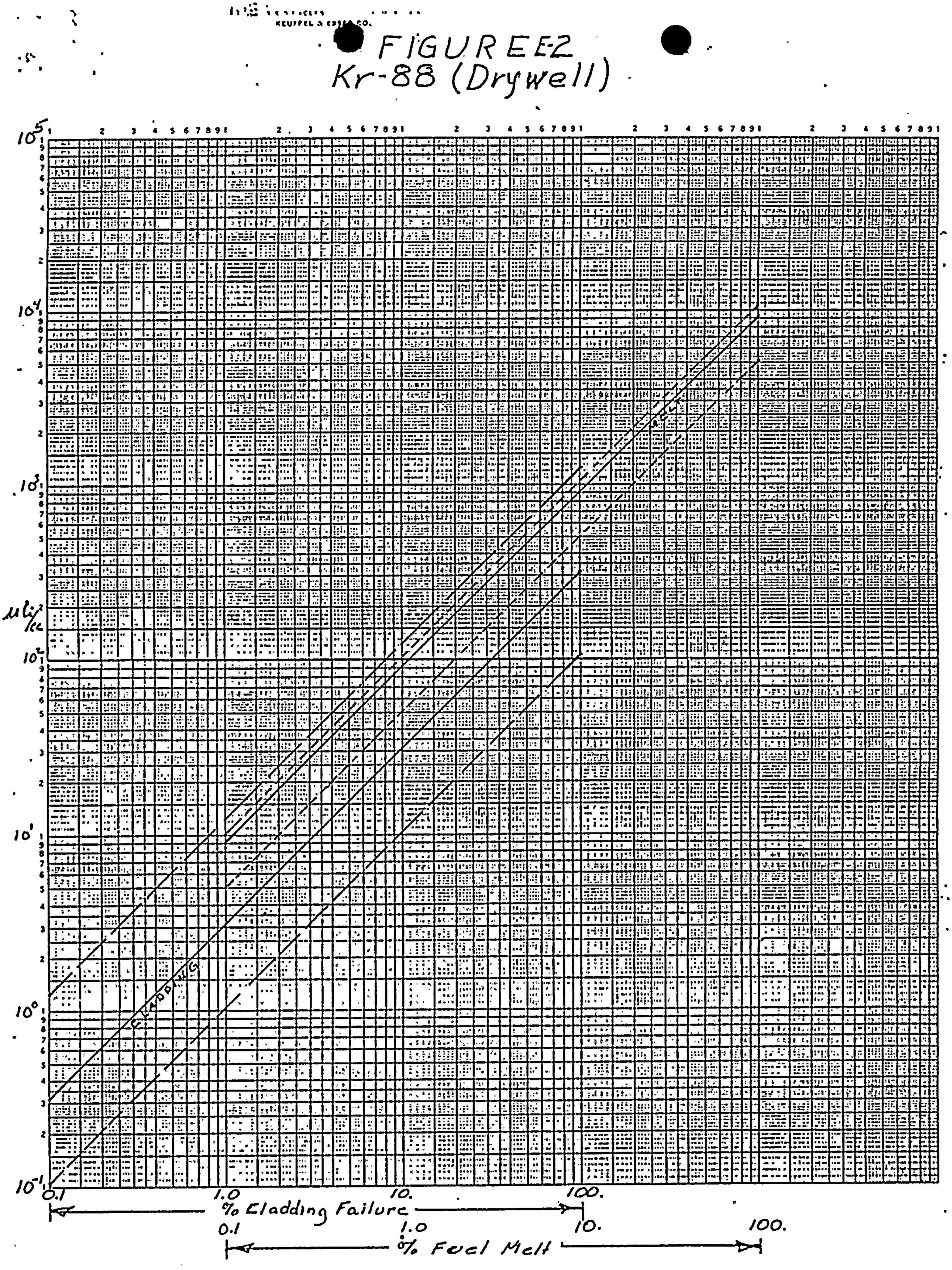


FIGURE E3
Xe-133 (Drywell)

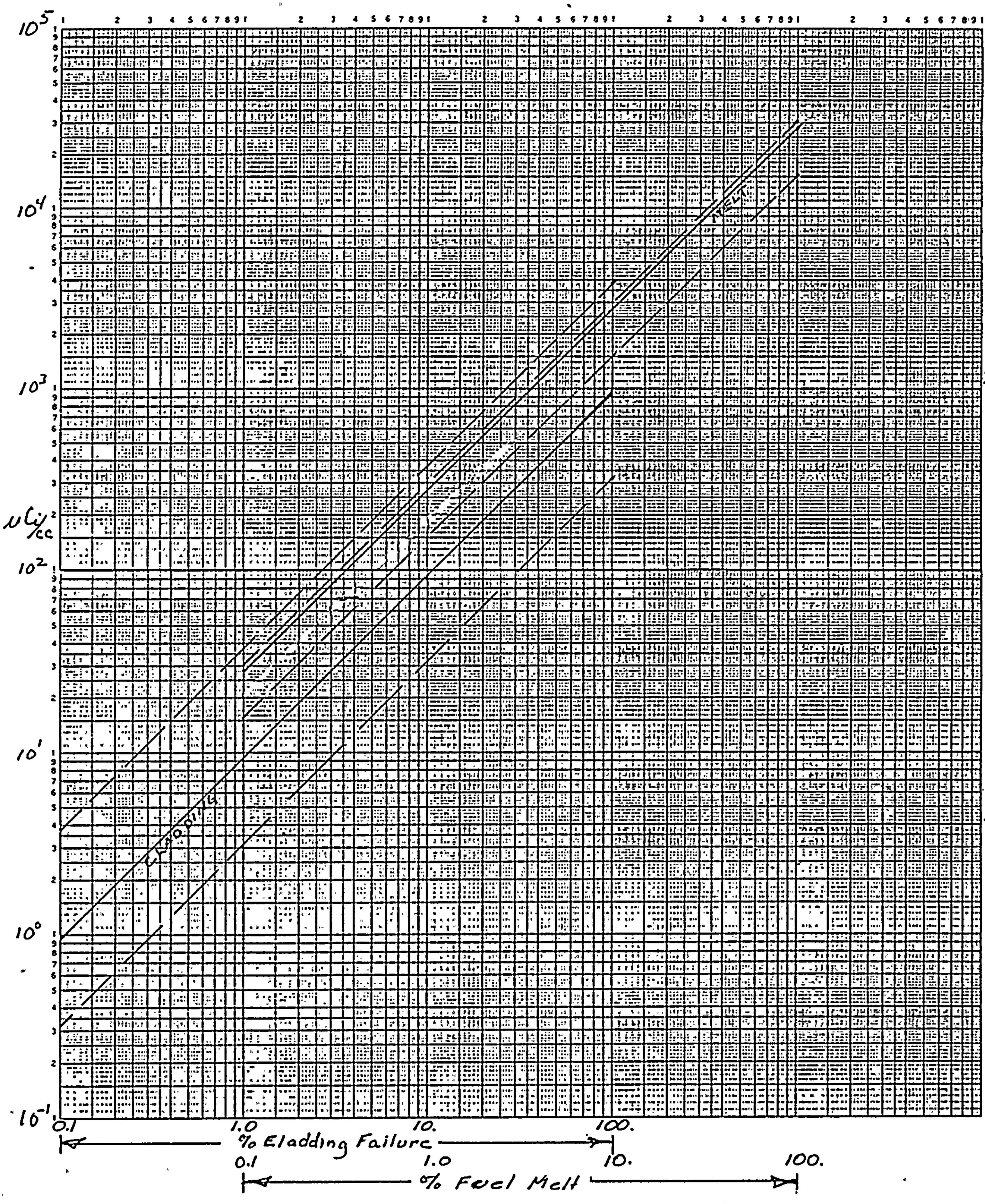


FIGURE E-4
Xe-135 (Drywell)

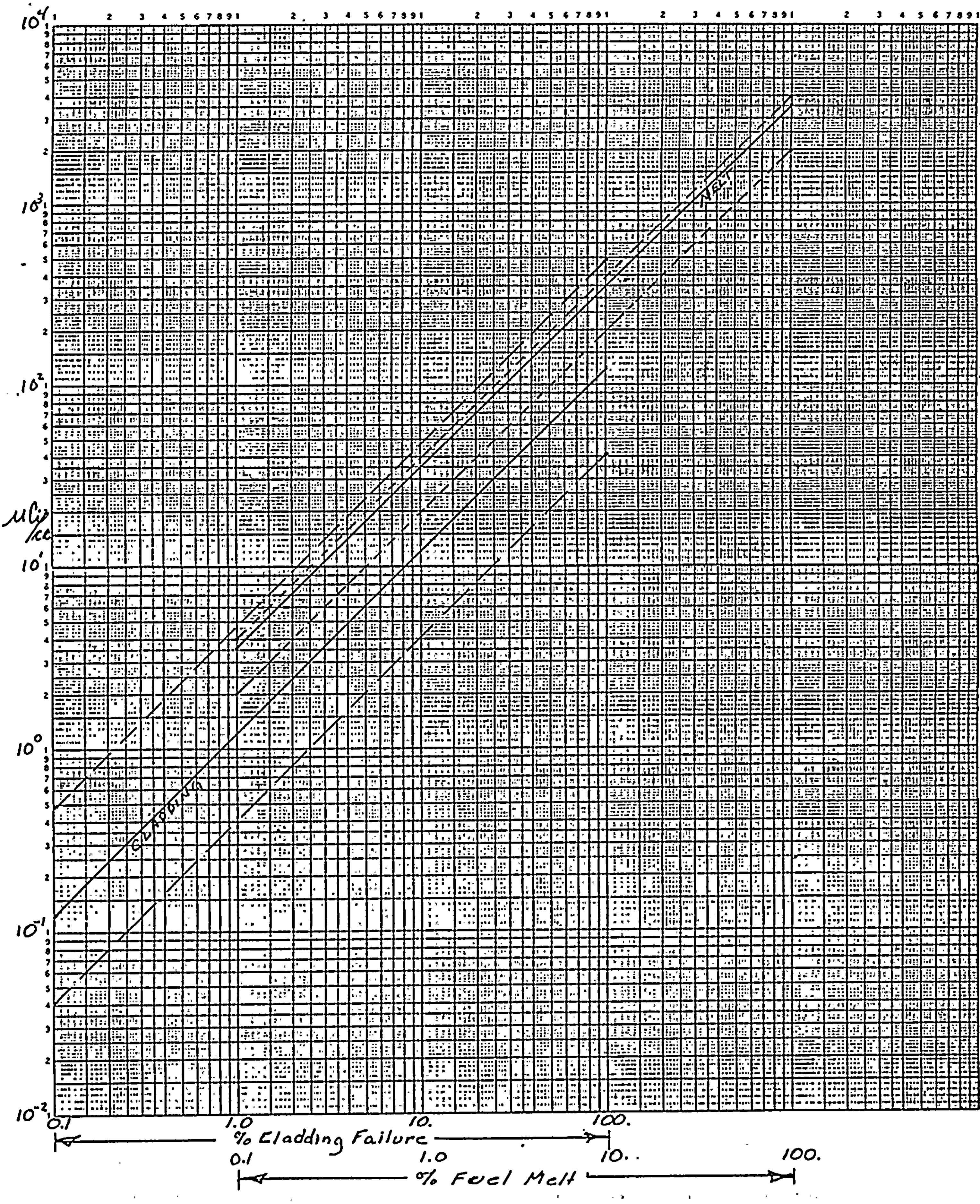


FIGURE-5
Kr-85 (Containment)

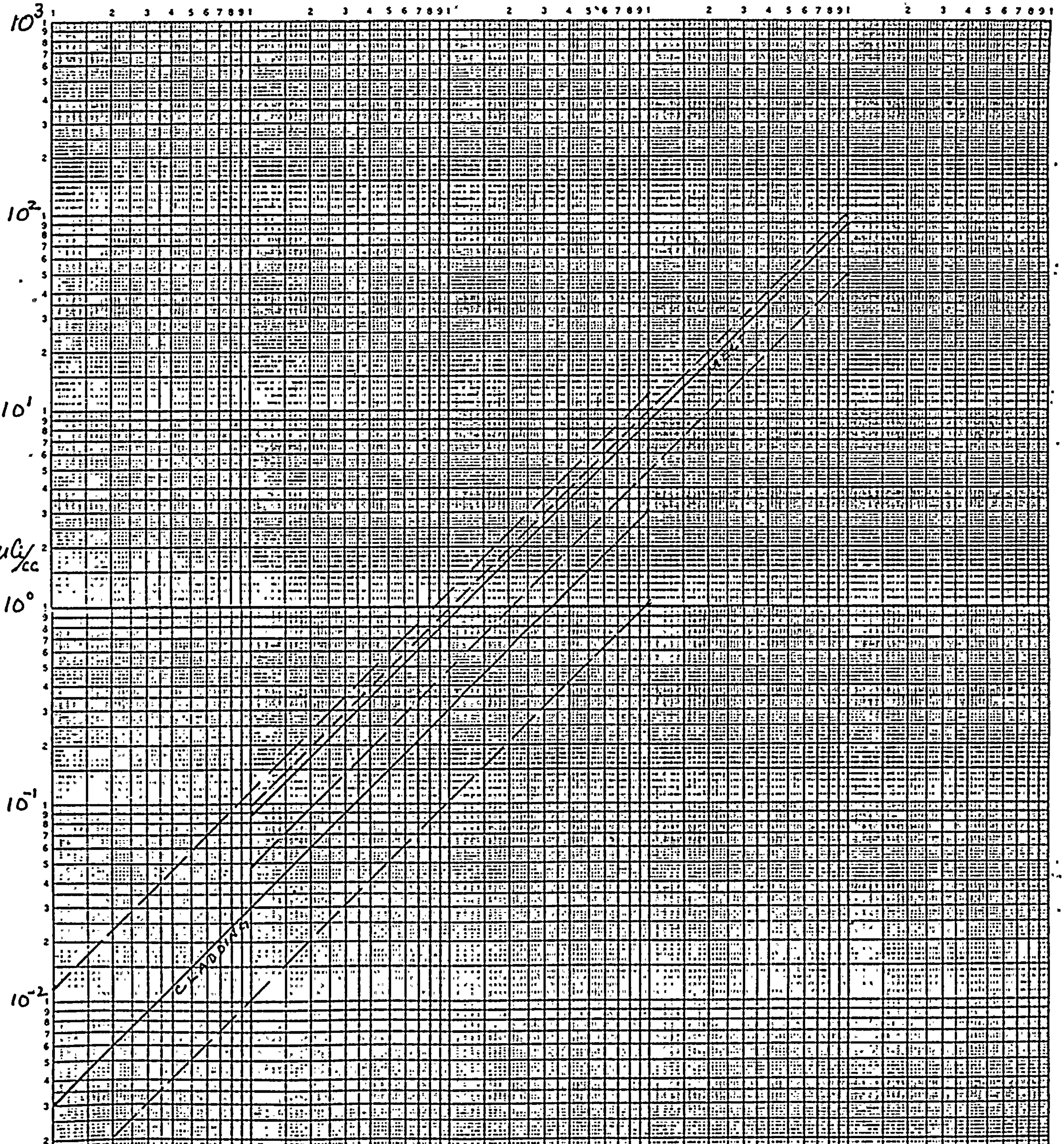


FIGURE-6
Kr-88 (Containment)

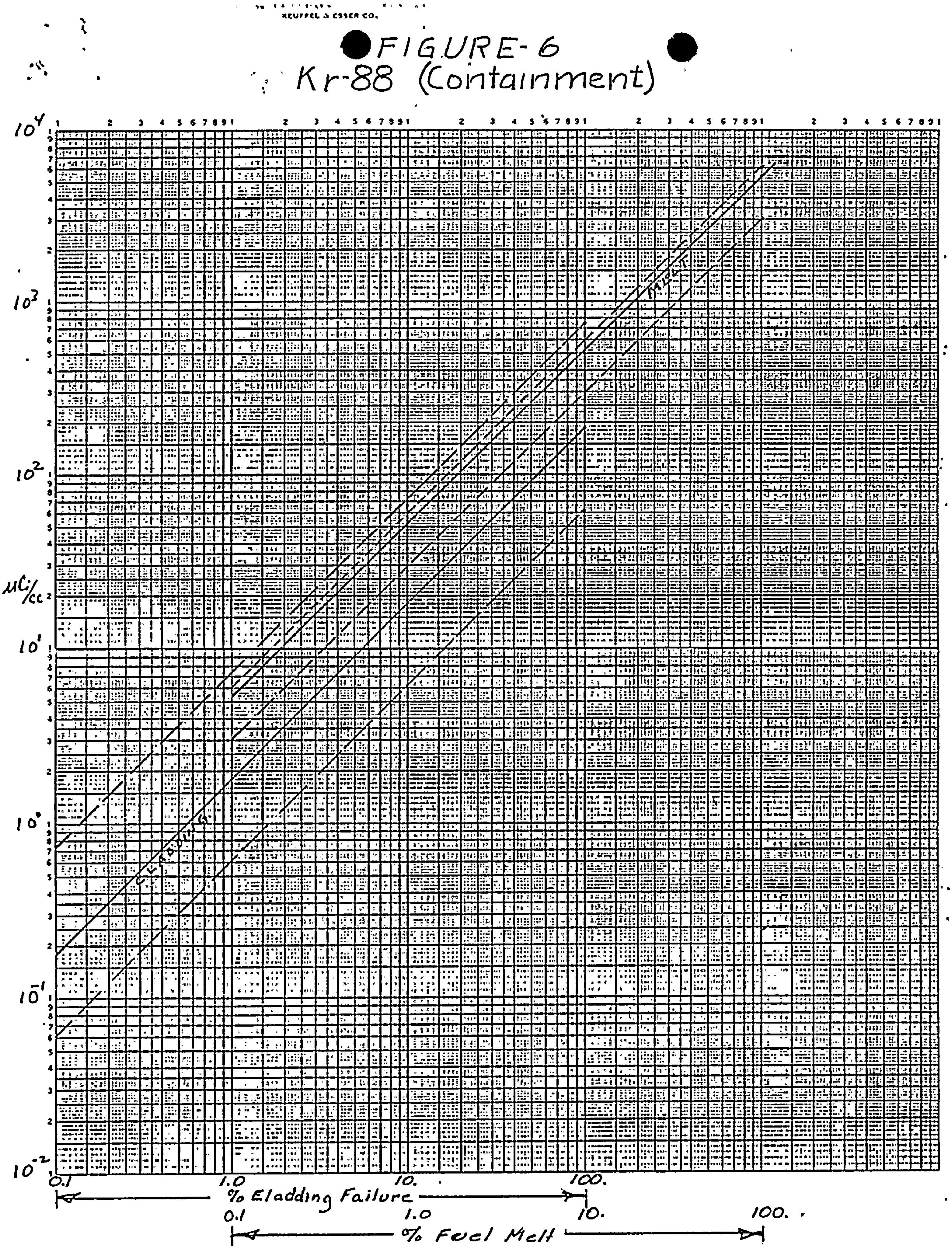


FIGURE-7
Xe-133 (Containment)

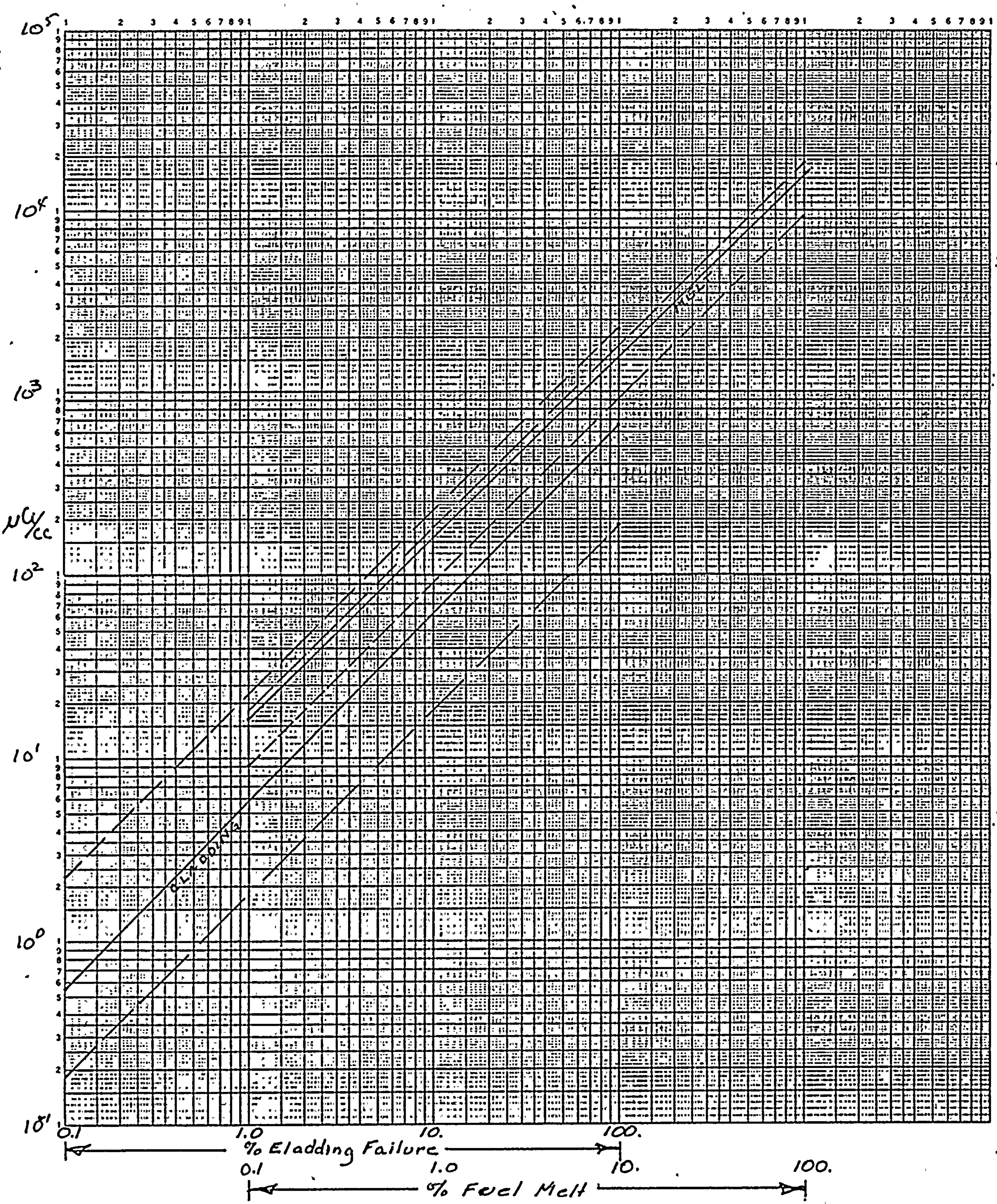


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FIGURE-8
Xe-135 (Containment)

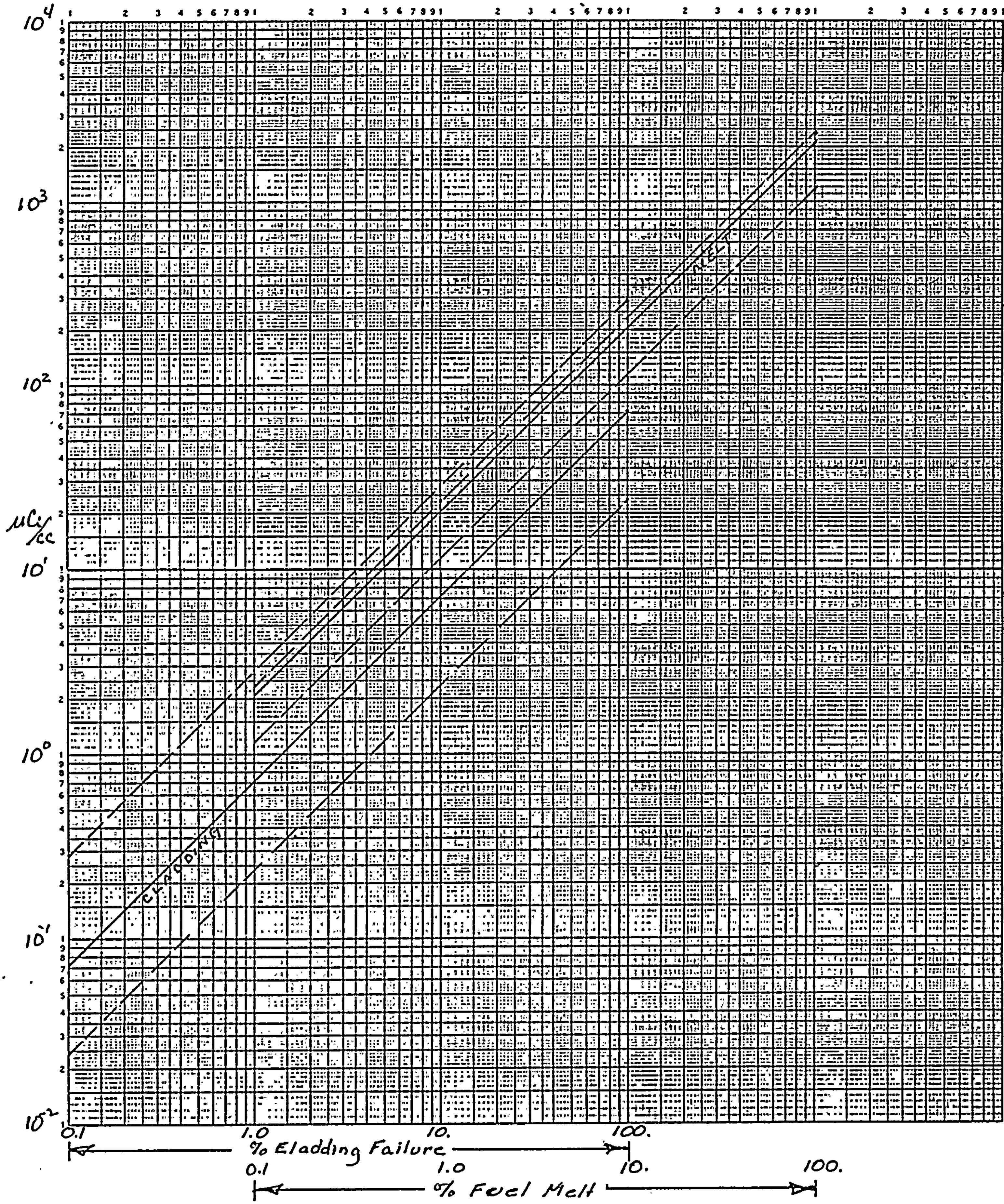

FIGURE-9

I-131 (Coolant System)

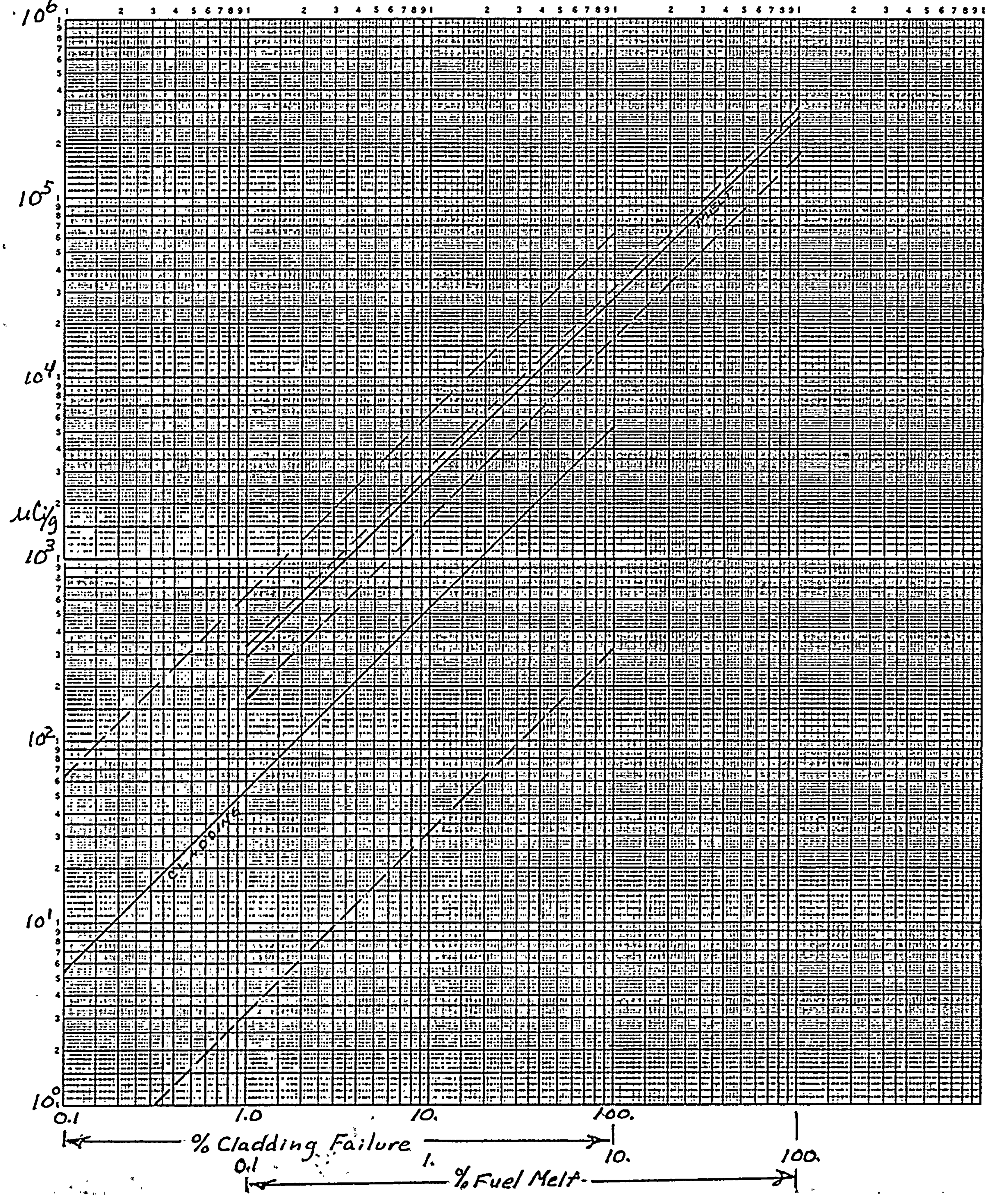

FIGURE-10

I-133 (Coolant System)

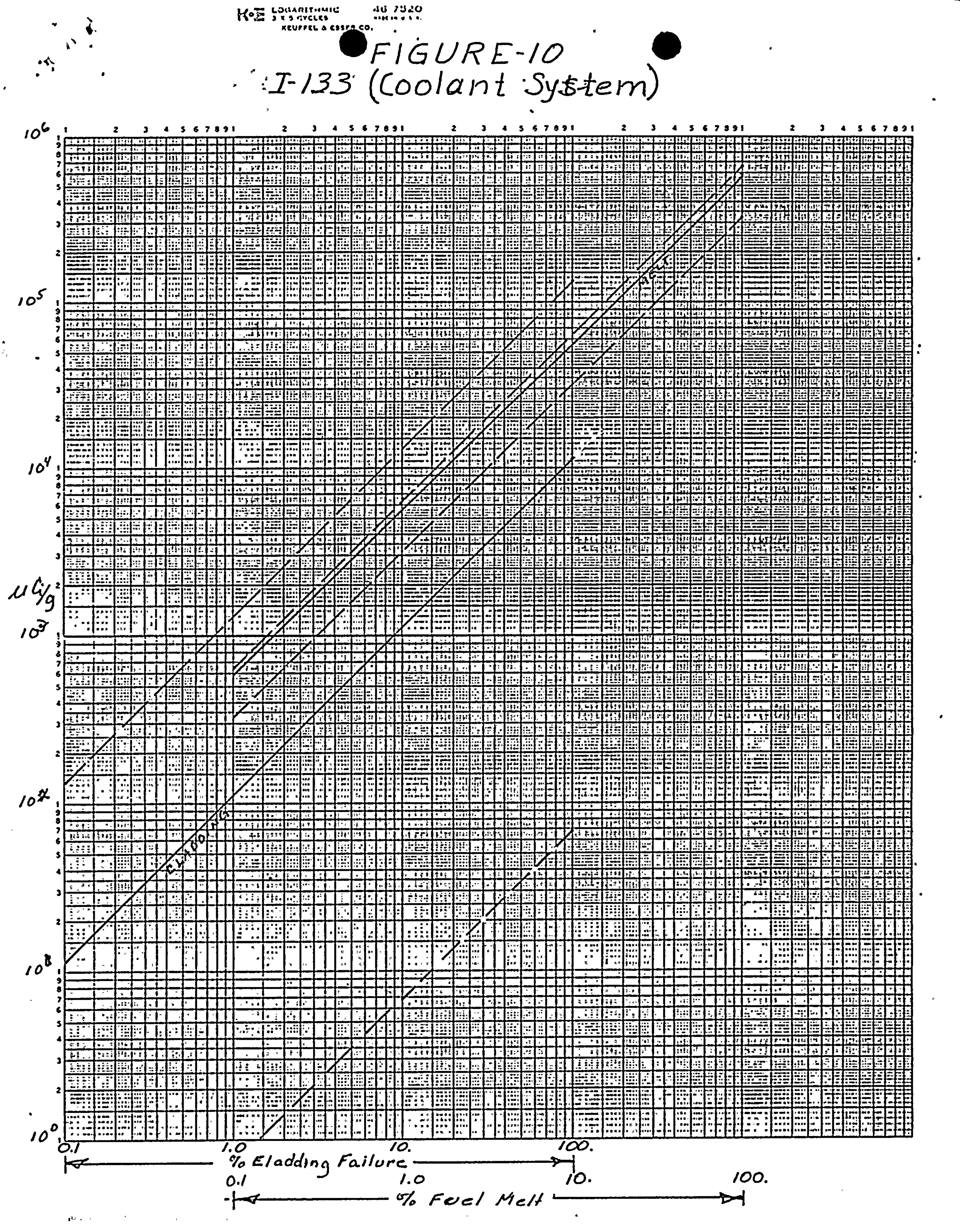

FIGURE - 11
Cs-134 (Coolant System)

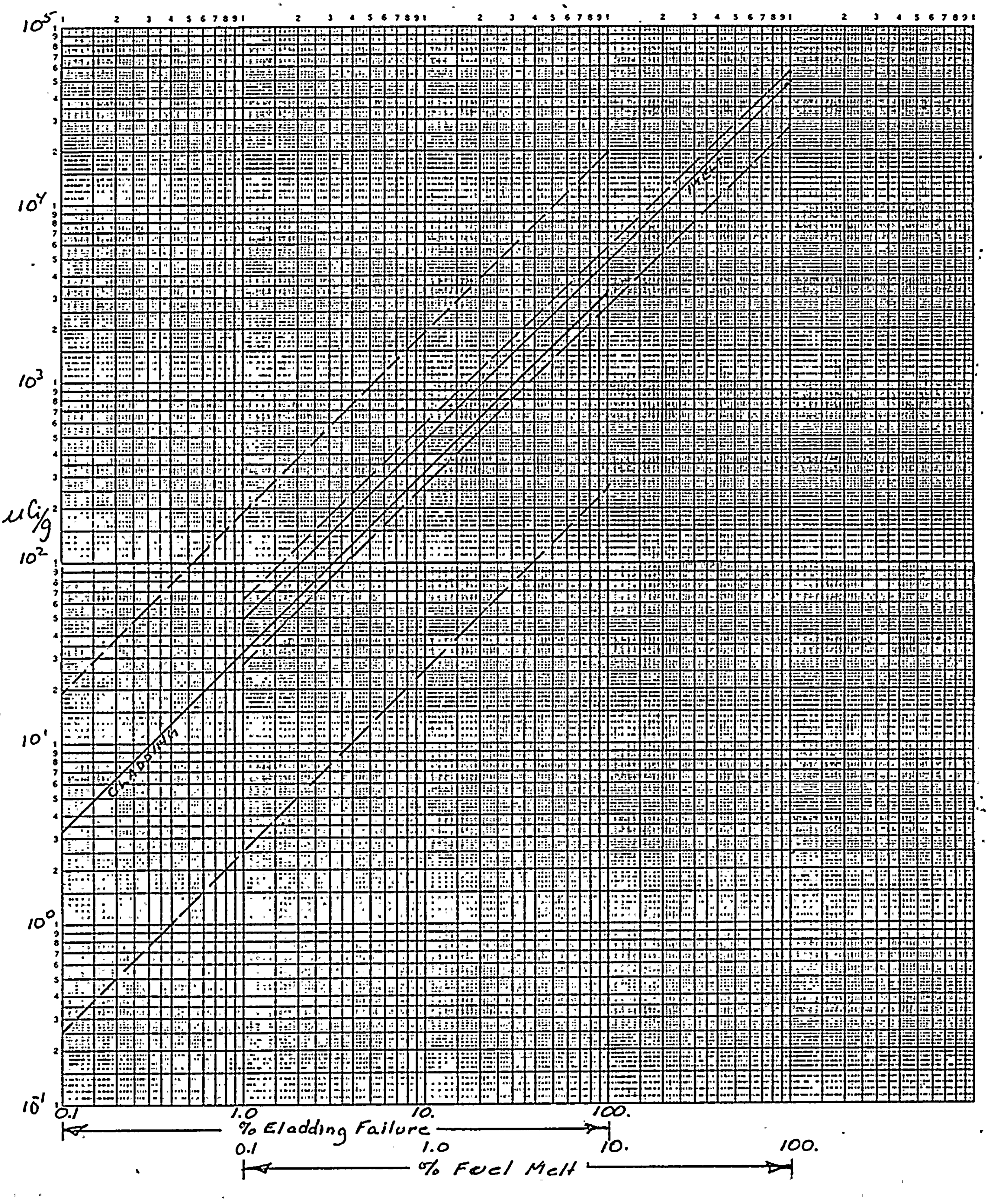

FIGURE-12
Cs-137 (Coolant System)

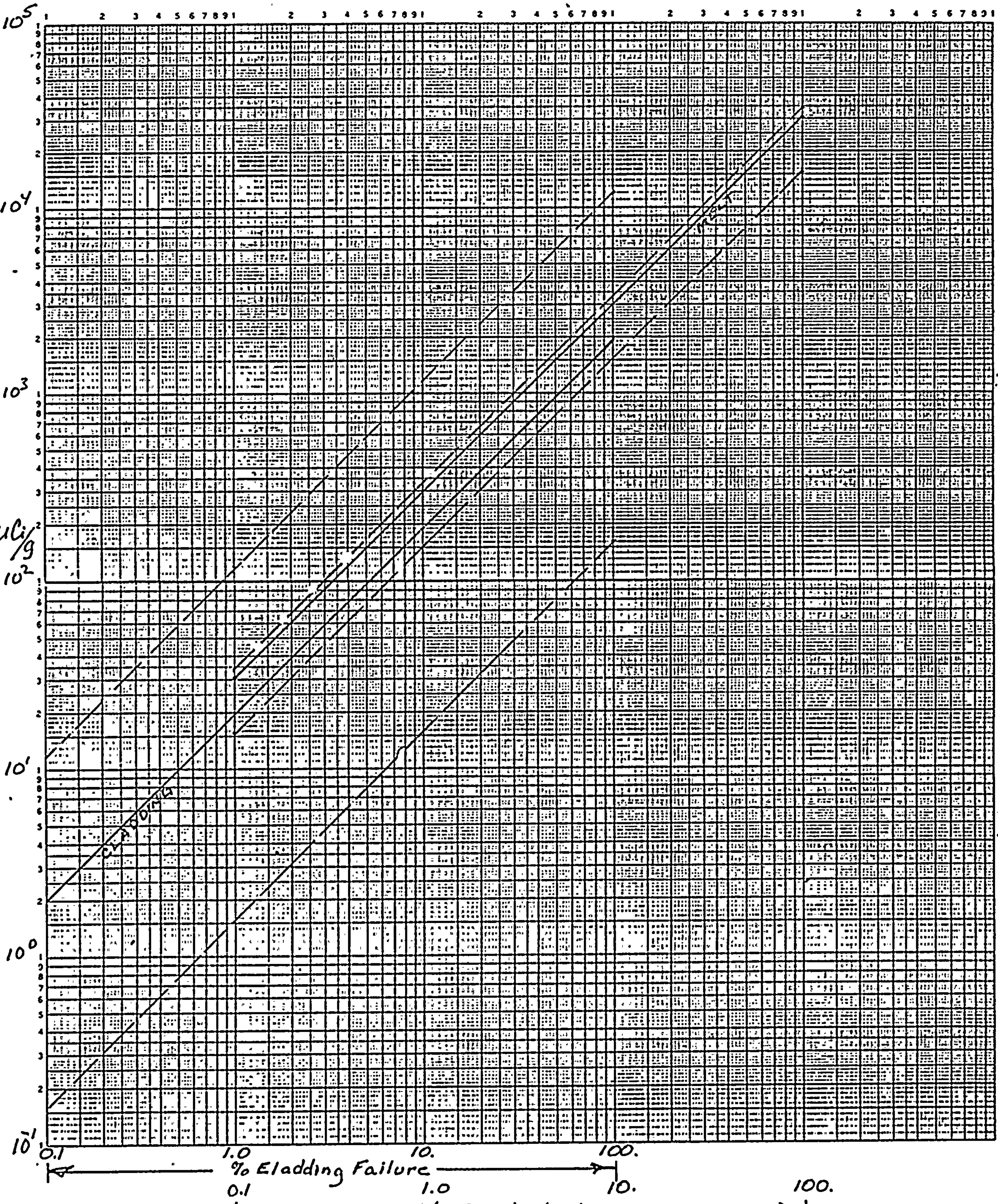

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FIGURE-13

I-131 (Coolant + Suppression Pool)

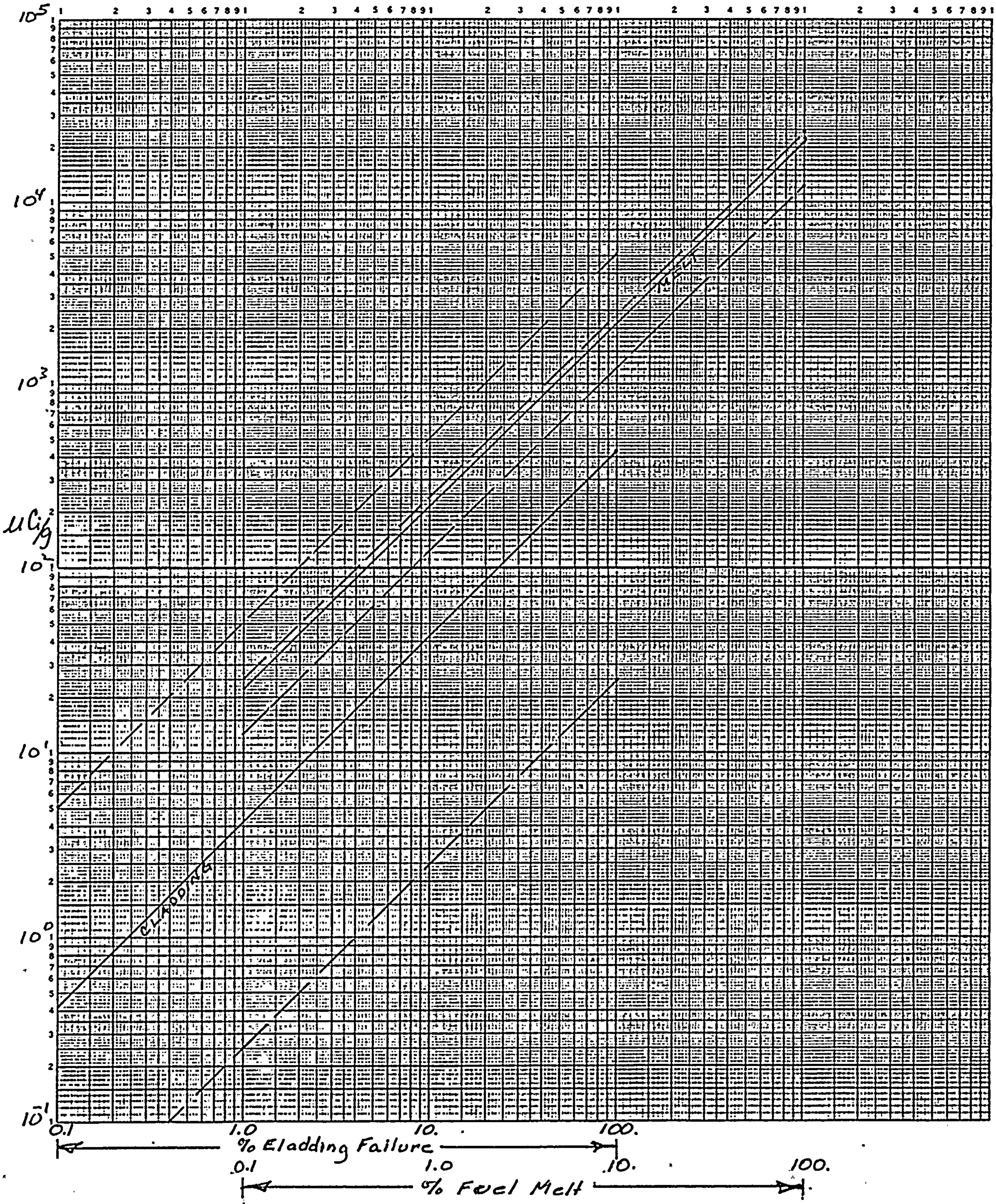

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FIGURE-14
I-133 (Coolant + Suppression Pool)

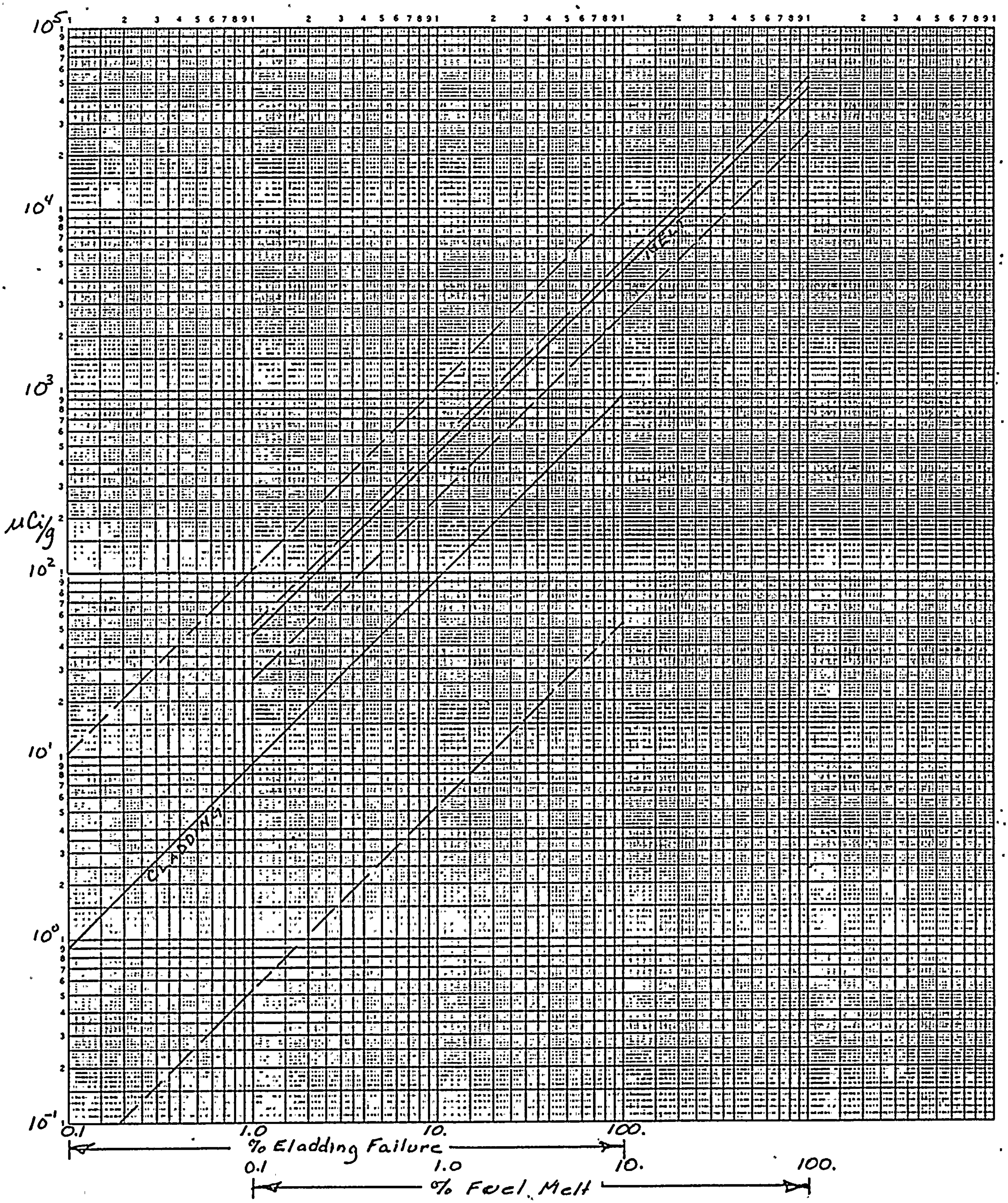


FIGURE E-15
Cs-134 (Coolant + Suppression Pool)

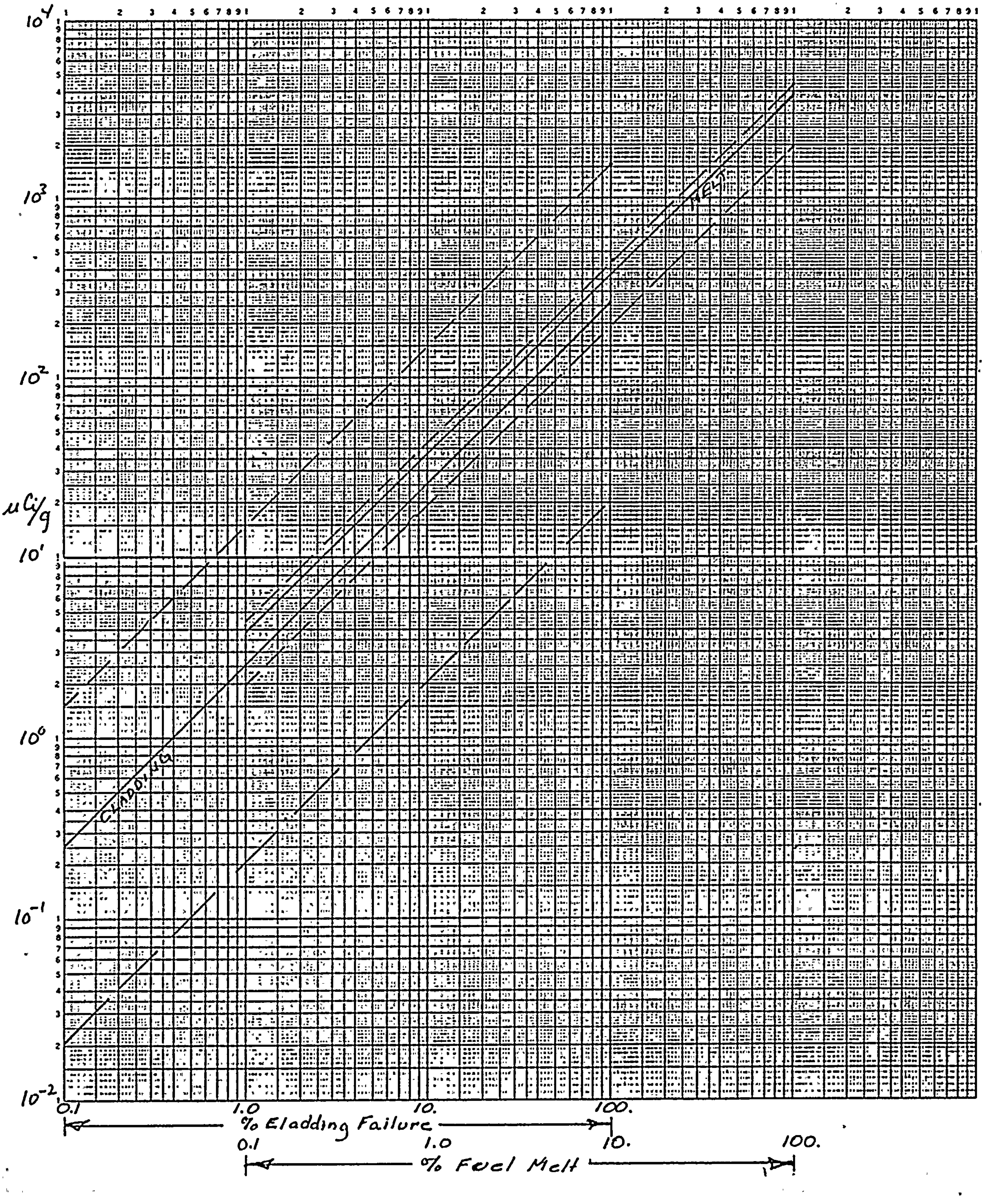
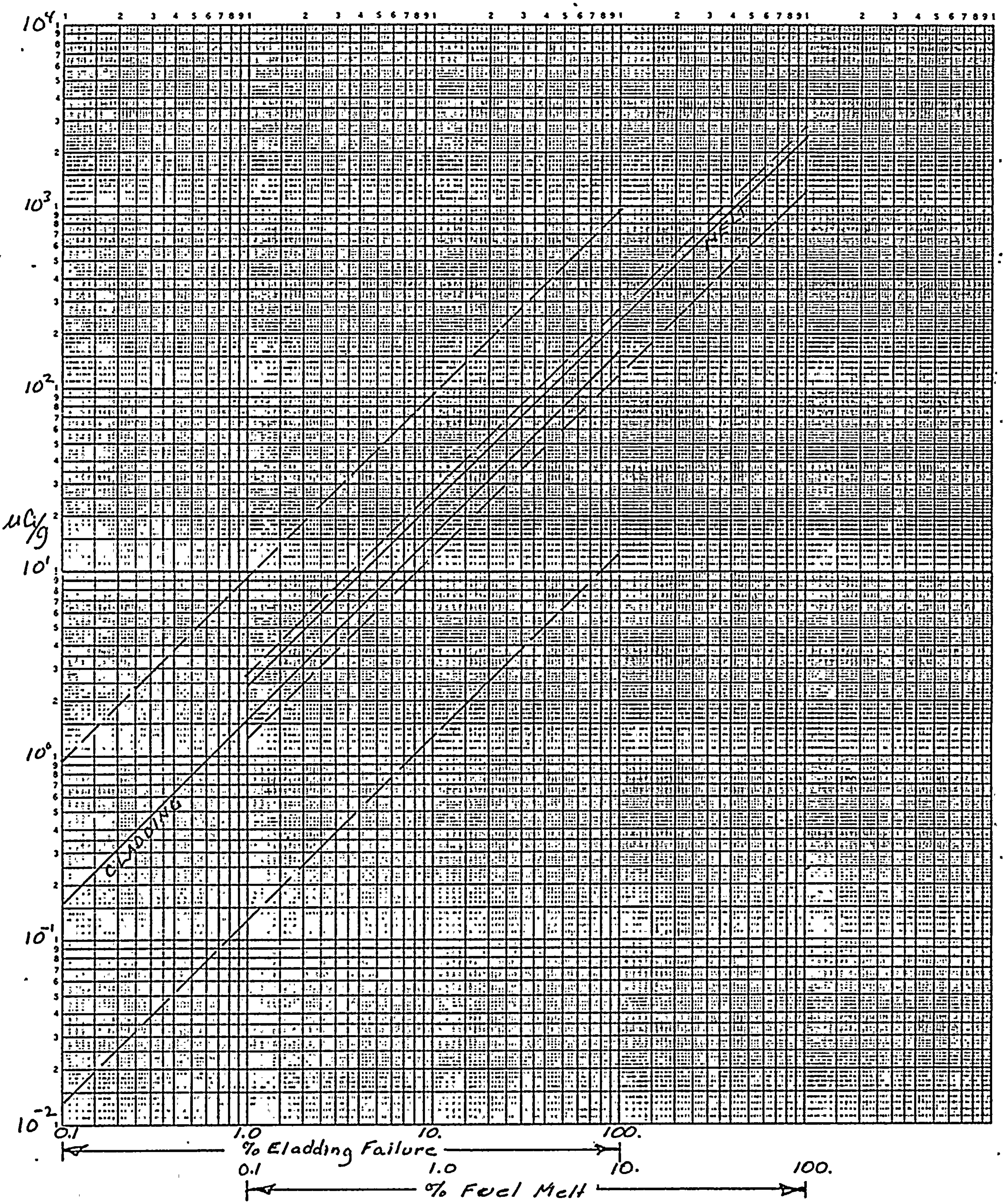


FIGURE E-16
Cs-137 (Coolant + Suppression Pool)



APPENDIX F

If a cladding failure condition exists then only the fuel gap release would be expected. If there is fuel melting then the core inventory release would be expected. The ratios of specific short lived noble gas isotopes to Xe-133 and of specific short lived iodines to I-131 will be significantly different for each condition. The expected ratios are given below in Table F-1. A comparison of actual measured ratios to these values will assist in determining which condition exists.

Table F-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 m	0.233	0.0234
Kr-88	2.84 h	0.33	0.0495
Kr-85m	4.48 h	0.122	0.023
Xe-133	5.25 d	1.0*	1.0
I-134	52.6 m	2.3	0.155
I-132	2.3 h	1.46	0.127
I-135	6.61 h	1.97	0.364
I-133	20.8 h	2.09	0.685
I-131	8.04 d	1.0*	1.0*

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

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

[illegible]

[illegible]

[illegible]

[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]

[illegible]

[illegible]