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NUCLEAR BUSINESS UNIT
ONSITE IMPLEMENTING PROCEDURES
April 10, 2000

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CHANGE PAGES FOR
REVISION #02

The Table of Contents forms a general guide to the current revision of each section of the Onsite EPIPs. The changes that are made in this TOC Revision #02 are shown below. Please check that your revision packet is complete and remove the outdated material listed below:

ADD			REMOVE		
Page	Description	Rev.	Page	Description	Rev.
ALL	TOC	02	ALL	TOC	01
ALL	HC.EP-EP.ZZ-205	01	ALL	HC.EP-EP.ZZ-205	00

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PSEG NUCLEAR EMERGENCY PLAN
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STATION PROCEDURES

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HC.EP-EP.ZZ-0205 (Q) Rev. 001

TSC – POST ACCIDENT CORE DAMAGE ASSESSMENT

USE CATEGORY: II

REVISION SUMMARY:

Corrected typo found on page 2 of 3 of Attachment 6. Value of 1.03E+8 was changed to the correct value of 3.03E+8.

No other changes have been made to the procedure.

PSE&G CONTROL COPY # EP1P059

IMPLEMENTATION REQUIREMENTS

Effective Date: April 10, 2000

APPROVED: Raymond Bace (R. Bace) for D. Miller — Manager - EP & IT — 04/06/2000 Date

APPROVED: ______ Vice President - Operations — ______ Date

TSC – POST ACCIDENT CORE DAMAGE ASSESSMENT

TABLE OF CONTENTS

ATTACHMENTS

1.0 PURPOSE

This procedure provides guidance for core damage assessment after an ALERT or higher level of emergency has been declared with the reactor shut down.

2.0 PREREQUISITES

2.1 Prerequisites To Be Followed Prior To Implementing This Procedure

Implement this procedure at:

- The discretion of Nuclear Fuels Engineer (NFE).
- Upon staffing of your Emergency Response Facility.

3.0 PRECAUTIONS AND LIMITATIONS

3.1 Precautions

3.1.1 It is recommended that initials be used in the place keeping sign-offs, instead of checkmarks, if more than one person may implement this procedure.

3.1.2 Personnel who implement this procedure shall be trained and qualified IAW the Emergency Plan.

3.2 Limitations

3.2.1 The core damage assessment methodology does not account for fission product spiking.

3.2.2 The core damage assessment methodology assumes reactor coolant cleanup systems are isolated.

3.2.3 Measurement of Cs-137 and Kr-85 activities may not be possible until shorter livec isotopes have decayed.

3.2.4 Clad damage of less than 1% is not considered to be a loss of the fuel cladding boundary.

4.0 EQUIPMENT REQUIRED

As provided in the Emergency Response Facility.

5.0 **PROCEDURE**

NOTE:

Due to the multiple and, at times, unpredictable failure mechanism associated with core damage this procedure has been developed to provide GUIDANCE for Core Damage Assessment. The sequence and extent of procedure performance should be based on the knowledge and experience of the Nuclear Fuels Engineer.

5.1 **Nuclear Fuels Engineer Should Perform the Following to Initiate Core Damage Assessment and CDA Sample Results:**

5.1.1 PERFORM HCGS plant-specific calculations and estimations of the types and extent of reactor fuel damage utilizing the guidance of this procedure. [CD-385Y] [CD-548X] ______

5.1.2 PROVIDE recommendations to the Radiological Assessment Coordinator (RAC) to initiate post accident radionuclide samples and review all requests for post accident radionuclide samples for the purposes of core damage assessment. [CD-443D] ______

5.1.3 OBTAIN the post accident radionuclide samples with consideration as to how representative the sample will be of the core condition. ______

5.1.4 DETERMINE need and frequency for post accident radionuclide samples shall be determined with consideration of the application of the assessment – Accident Classification, Accident Mitigation, Source Term Assessment, Core Damage Assessment Only. ______

5.1.5 Recommend to the RAC post accident sampling system sample points based upon reactor condition or event type. ______

A. SELECTION of Liquid Sample Point

NOTE

Residual Heat Removal (RHR) samples: If RHR is in the Low Pressure Coolant Injection (LPCI) or Suppression Pool Cooling modes, it should be operating an estimated 30 minutes minimum prior to sampling to ensure a representative sample. [CD-384Y]
PASS Reactor Jet Pump samples: At power levels < 1% of rated power, a representative sample may be obtained by raising reactor water level 18 inches to fully flood the moisture separators thus providing a thermally induced recirculation flow path for mixing.

LIQUID SAMPLE

LOCATION	TYPE	MODE
Reactor (Jet Pump)	PASS	Reactor Pressurized
Reactor (RHR-LPCI/Shutdown Cooling)	PASS	Reactor Depressurized
Suppression Pool (RHR) *	PASS	RHR in Suppression Pool Cooling *

* [CD-384Y]

GAS SAMPLE

EVENT	SAMPLE LOCATION
Non – Break	Torus Atmosphere
Small Break	Prior to Depressurization: Drywell Atmosphere After Depressurization: Torus Atmosphere
Large Break (in Drywell)	Drywell Atmosphere
Large Break (outside Drywell)	Torus Atmosphere

B. RECORD on Attachment 1 the current time, selected sample point, the desired frequency of sampling and the basis for the selection and frequency. ______

C. PROVIDE a copy of Attachment 1 to the RAC and the Technical Support Team Leader (TSTL). ______

5.1.6 ESTIMATE the type and extent of core damage based on the Drywell Atmosphere Post Accident (DAPA) Radiation Monitor Reading ______

5.1.7 IF the Drywell Atmosphere Post Accident Monitor has been declared inoperable by Operations, THEN GO TO step 5.2. ______

5.1.8 OBTAIN and record on Attachment 2, DRYWELL ATMOSPHERE POST ACCIDENT (DAPA) MONITOR A AND B READING (R/HR), the time of the reading and the time of reactor shutdown. ______

NOTE

DAPA monitor A and B provide indication for two different locations in the Drywell.

If adverse conditions exist in the Drywell (average Drywell air temperature greater than or equal to 245°F) validate with the Radiological Assessment Coordinator that EPIP 302H, Attachment 5, DAPA CORRECTION CALCULATIONS has been utilized.

5.1.9 DETERMINE the percent of fuel inventory airborne in using Attachment 2. Record the result on Attachment 17. ______

5.2 Estimating the Type and Extent of Core Damage Based on the Drywell Atmosphere Post Accident (DAPA) Equivalent Calculation

5.2.1 GO TO step 5.1.5 if the DAPA monitor is operable.

5.2.2 INFORM the TSTL of the need to determine drywell atmosphere radiation levels without the DAPA monitor. ______

5.2.3 REQUEST from the Radiological Assessment Coordinator a "Contact Dose Rate" at the Drywell Personnel Airlock and a Particulate, Iodine, and Noble Gas air Sample of the 120' El. Rx. Bldg. For the purposes of determining a DAPA EQUIVALENT READING. Calculate a "DAPA EQUIVALENT" value and document the value on Attachment 2 as a DAPA EQUIVALENT in the following manner: ______

EQUIV = 100 x (CDR – (20 x (NUGC))

WHERE: EQUIV = DAPA Equivalent (R/HR) for use in Attachment 2
(if CDR = normal bkg then CDR = 0)
CDR = Contact Dose Rate (R/HR)
NGC = Nobel Gas Concentration (μCi/cc)
(if NGC is < 1E-04 (μCi/cc then NGC = 0)

5.2.4 DETERMINE the percent of fuel inventory airborne by using Attachment 2. Record the result on Attachment 17. ______

5.3 **Determining the Percent of Metal-Water Reaction from the Hydrogen Concentration in the Primary Containment Free Volume.**

5.3.1 OBTAIN the hydrogen concentration in the primary containment from the Hydrogen-Oxygen Analyzer System or from the post accident sampling system and record it on Attachment 3. ______

5.3.2 RECORD the time of the reading or sample and the sample point on Attachment 3. ______

5.3.3 RECORD on Attachment 3 any drywell venting or hydrogen recombiner operation. ______

5.3.4 DETERMINE the percent metal-water reaction by using Attachment 3. Record the result on Attachment 17. ______

5.4 **Estimating the Type and Percent of Core Damage From Fission Product Concentrations.**

5.4.1 IF a liquid sample has been selected as identified on Attachment 1, obtain from the Chemistry Supervisor in the TSC the concentration of I-131, I-132, I-133, I-134, I-135, Cs-134, Cs-137, sample point, sampling time, sample analysis time, type of decay correction performed and the time of final reactor shutdown. Record the information on Attachment 4. ______

5.4.2 IF a gas sample has been selected as identified on Attachment 1, obtain from the Chemistry Supervisor in the TSC the concentration of Kr-85m, Kr-85, Kr-87, Kr-88, Xe-133, Xe-135, sample point, sampling time, sample analysis time, type of decay correction performed, and the time of final reactor shutdown. Record the information on Attachment 4. ______

5.4.3 CALCULATE the pressure/temperature corrected fission product concentrations for gas sample radioisotopes as per Attachment 4. ______

NOTE
Pressure/temperature corrections will not be necessary if the corrections have been performed by the Chemistry Department.

5.4.4 CALCULATE the decay corrected fission product concentrations as per Attachment 4 and record the results on Attachment 4. ______

> **NOTE**
>
> Decay corrections will not be necessary if performed by the Chemistry Department.

5.4.5 CALCULATE the fission product inventory correction factors (F_i) as per Attachment 5. ______

5.4.6 CALCULATE the normalized concentrations of the fission products (C_{wn}) as per Attachment 6. ______

5.5 **Utilizing the Normalized Concentrations**

5.5.1 Following the instructions on Attachment 6 and Attachments 7 through 15 estimate the percent cladding failure and percent fuel melting. ______

5.5.2 Record the results on Attachment 17. ______

> **NOTE**
>
> The lines on the graphs are set up in the following manner:
>
> Upper Dashed Line – maximum fission product release for a given fuel condition.
> Lower Dashed Line – minimum fission product release for a given fuel condition.
> Center Solid Line – nominal fission product release for a given fuel condition.

5.6 **Estimating Release Source (Gap or Fuel Pellet) From the Isotopic Ratios.**

5.6.1 CALCULATE the isotopic ratios as per Attachment 16. ______

5.6.2 COMPARE the calculated isotopic ratios to the values listed in the table on Attachment 16 to estimate the release source. Record the results on Attachment 17. ______

5.7 **Determine If Less Volatile Fission Products are Present in the Reactor Coolant.**

5.7.1 IF the less volatile fission products, such as Sr, Ba, La, or Ru (either soluble or insoluble), are found to have unusually high concentrations in the reactor coolant some degree of fuel melting may be inferred. ______

5.7.2 RECORD observations of less volatile fission products on Attachment 17. ______

5.8 Estimating If an Interruption of Adequate Core Cooling Has Occurred.

5.8.1 OBTAIN a history of the reactor vessel water level from the initiation of the accident from SPDS or the VAX LA120. ______

5.8.2 DETERMINE if the top of active fuel (TAF) has been uncovered. ______

5.8.3 RECORD the level history, duration of level below the TAF and an estimate of cooling adequacy on Attachment 17. ______

NOTE

Significant or core-wide damage is not expected unless the TAF has been uncovered. Core-wide clad damage can occur within 30 minutes of uncovering the fuel. However, unless level is below the bottom of the active fuel, boiling heat transfer will provide cooling and significantly extend the duration that a partial uncovering can be withstood without significant core damage.

5.9 Performing an Assessment of the Type and Extent of Core Damage Based Upon All Available Indicators

5.9.1 CLASSIFY the type and extent of core damage relative to the following matrix. ______

Degree of Core Damage	Minor (<10%)	Intermediate (10% - 50%)	Major (>50%)
None (<1% clad)	1	1	1
Clad Failure	2	3	4
Fuel Overheat	5	6	7
Fuel Melt	8	9	10

NOTE

The primary indicator for the damage assessment is the estimate of damage based on the post accident radionuclide sample data as utilized in Section 5.5.

5.9.2 EVALUATE the other indicators or parameters to corroborate and further refine the assessment as determined in section 5.5. ______

5.9.3 REQUEST that the TSTL INITIATE appropriate confirmation of accuracy if conflicting indications are identified. ______

5.9.4 RECORD the assessment and bases on Attachment 17. ______

5.10 **Reporting the Results of the Assessment and Recommending Further Actions.**

5.10.1 REPORT the results to the TSTL for dissemination to the TSS and the RAC. ______

5.10.2 REVIEW the current accident status in order to make recommendations for further actions to refine or continue the assessment. ______

5.10.3 RE-ENTER the procedure as appropriate. ______

6.0 RECORDS

Return completed procedure, original copies to the Manager – CA, E: &IT.

7.0 REFERENCES

7.1 References

7.1.1 General Electric Document, NEDO-22215 82NEDO90, Procedures for the Determination of the Extent of Core Damage Under Accident Conditions, August 1982.

7.1.2 General Electric Document, C&RE Transmittal, RPE 81CL01, November 1981

7.1.3 PSE&G Radiation Protection/Chemistry Services File NRP-88-0048, Preplanned Alternate Monitoring Methods for the DAPA Monitoring System, March 3, 1988.

7.1.4 Hope Creek UFSAR 1.14.1.49.2

7.2 Cross References

7.2.1 EPIP 302H/NC.EP-EP.ZZ-0302(Q), Radiation Assessment Coordinator Response

7.2.2 NBU Emergency Plan

7.3 **Closing Documents**

7.3.1 Hope Creek CD-443D

7.3.2 Hope Creek CD-384Y

7.3.3 Hope Creek CD-385Y

7.3.4 Hope Creek CD-548X.

ATTACHMENT 1
Page 1 of 1
POST ACCIDENT RADIONUCLIDE SAMPLE REQUEST

Sample Request No. ______________________

Time of Request ______________________

Sample Point ______________________

Frequency ______________________

Bases ______________________

Comments ______________________

Sample Request No. ______________________

Time of Request ______________________

Sample Point ______________________

Frequency ______________________

Bases ______________________

Comments ______________________

ATTACHMENT 2
Page 1 of 2
DAPA MONITOR DOSE RATE TO FUEL INVENTORY AIRBORNE

1. Time of Reactor Shutdown ______________________ Date ____________

2. Time of DAPA reading: A ______________________ Date ____________

   B ______________________ Date ____________

3. Time after Shutdown, Hrs ______________________________________

4. Complete the following Table for each reading.

	DAPA EQUIV (Y/N)	Monitor (A/B)	Dose Rate (R/Hr)	Time after Shutdown (Hrs)	Fuel Inventory From Graph, Iinv (%)
1					
2					
3					
4					
5					
6					

5. Use the following as guidance for assessing the amount and type of core damage based on the DAPA readings.

Iinv	Approximate Source and Damage Estimate
100	100% TID*, 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 uncovery
3	3% TID, 100% WASH-1400 gap activity, major clad failure
1	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 assemblies
0.01	0.01% TID, 0.1% NRC gap, clad failure of ¾ fuel assembly

* TID = Technical Information Document #14844 source term
100% Noble Gas, 25% Halogens and 1% Solids

ATTACHMENT 2
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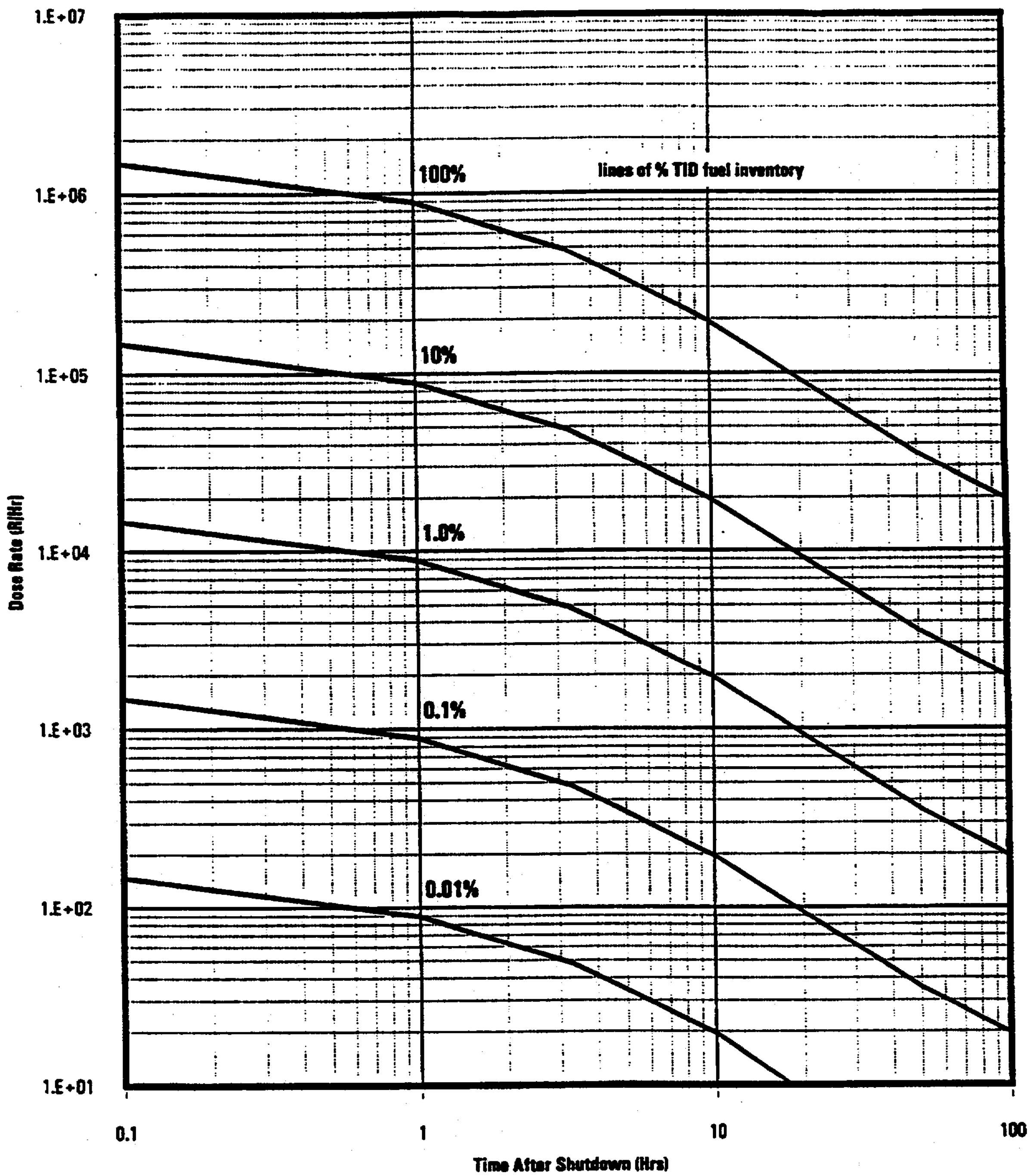

ATTACHMENT 3
Page 1 of 3
PRIMARY CONTAINMENT HYDROGEN CONCENTRATION TO % METAL – WATER REACTION

	Date	Time	System and Sample Point	H2 (%)	Metal-Water Reaction (%)
1					
2					
3					
4					
5					
6					

	Comments and Drywell Venting/Recombiner Operation Note
1	
2	
3	
4	
5	
6	

ATTACHMENT 3
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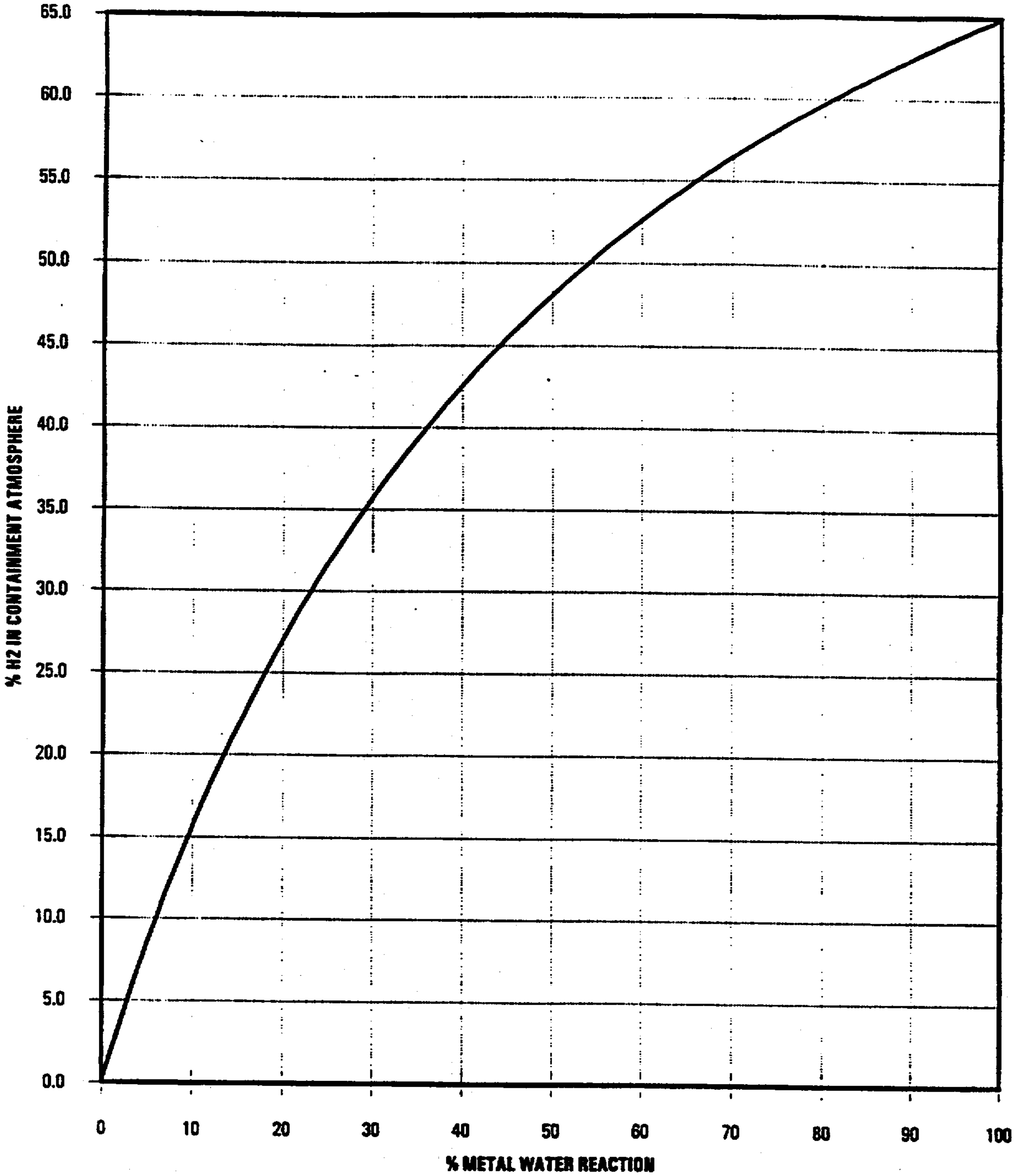

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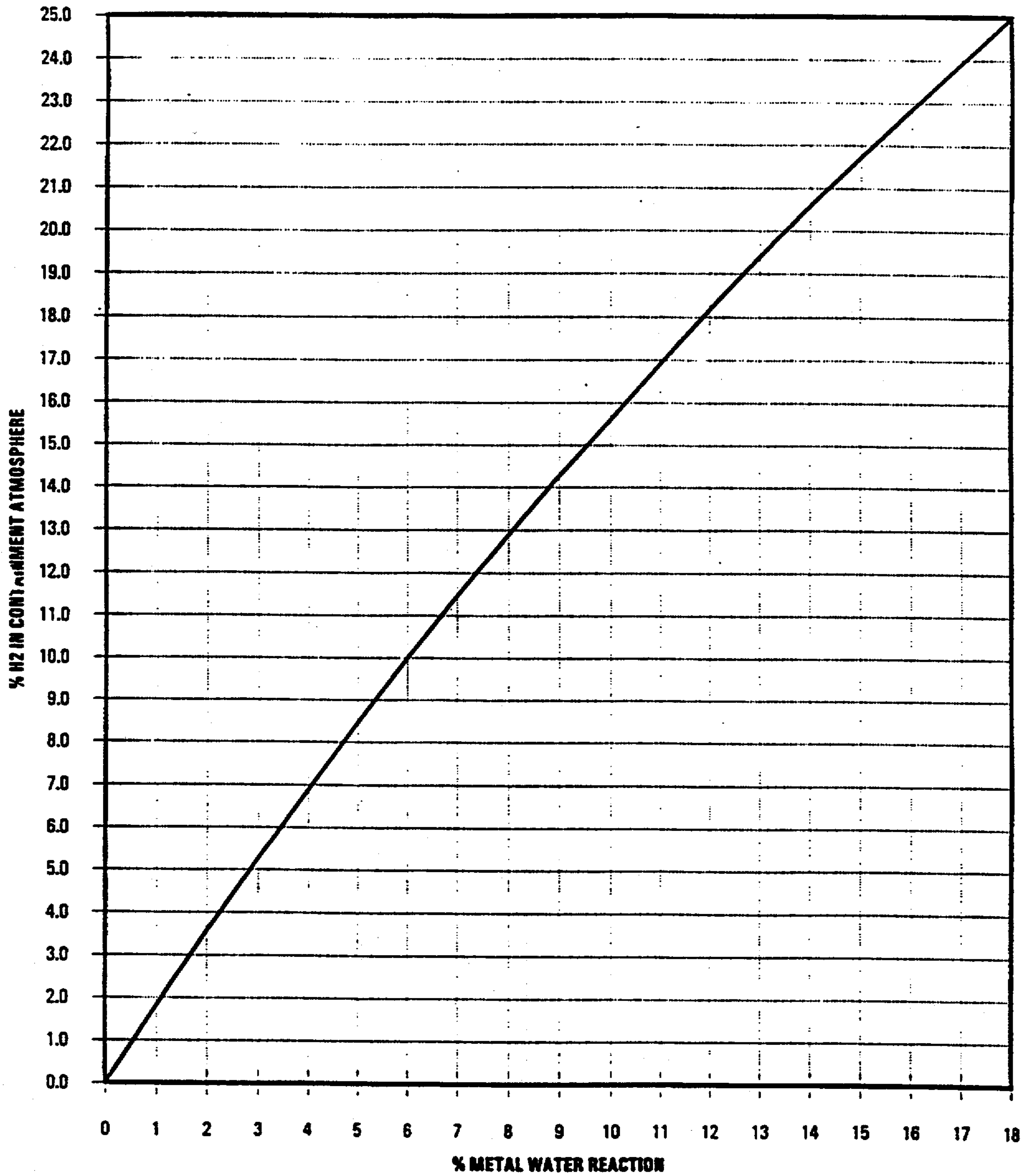

ATTACHMENT 4
Page 1 of 2
FISSION PRODUCT CONCENTRATIONS

Time of Reactor Trip or Shutdown ____________________

Sample No. ____________ Sample Time ______________

Sample Analysis Time ____________________________

Sample Type ____________ Sample Point ______________

Pressure/Temperature Correction ______________________________________

Sample Vial P1 ____________ psi T1 ____________ OK

Sample Point Environment P2 ____________ psi T2 ____________ OK

PTMULT = P2 T1/ P1 T2 = ______________ (PTMULT = 1.0 for liquid)

Decay Correction ____

DMULT = $e^{\lambda t}$

λ = decay constant of the isotope of interest (1/days)
t = time of decay (days)

NOTE

The time of decay must represent the elapsed time from reactor trip or shutdown to the sample analysis time.

NOTE

The decay correction must account for the activity decrease during the time period from reactor trip or shutdown to the sample analysis time.

ATTACHMENT 4
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	λ (1/day)	Sample Result (µCi/g)	P/T Corrected (µCi/g)	Decay Corrected (µCi/g)
I-131	8.621E-02			
I-132	7.23E+00			
I-133	7.998E-01			
I-134	1.898E+01			
I-135	2.517E+00			
Cs-134	9.219E-04			
Cs-137	6.294E-05			
Kr-85m	3.713E+00			
Kr-85	1.771E-04			
Kr-87	1.308E+01			
Kr-88	5.858E+00			
Xe-133	1.320E-01			
Xe-135	1.826E+00			

ATTACHMENT 5
Page 1 of 3
FISSION PRODUCT INVENTORY CORRECTION FACTORS

1. Calculate the inventory correction factor (Fi) for each fission product listed in steps 3 and 4 of Attachment 5 using the following:

   1.1 Bases:

   $$F_i = \frac{\text{reference inventory of isotope i in HCGS}}{\text{actual inventory of isotope i in HCGS}}$$

   1.2 If the total operating time for all batches is greater than or equal to the power correction time:

   $$F_i = \frac{3293\,(1-e^{-1095\lambda i})}{\Sigma_j\,[P_j(1-e^{-\lambda iTj})e^{-\lambda iT0}]}$$

Where:	λ_i	=	decay constant of isotope i (1/days)
	Σ_j	=	sum of the batches
	P_j^*	=	steady reactor power (total core power) operated in period j (MWt)
	T_j^*	=	duration of operating period j (days)
	T_0	=	time between the end of operating period j and time of the final reactor shutdown (days)

* For each time period, T_j^* the variation of steady reactor power, P_j^*, should be limited to $\pm$ 20%.

1.3 If the total operating time for any batch is less than the power correction time:

$$F_1 = \frac{3293(1-e^{-1095\lambda i})}{\Sigma_k \Sigma_j [BP_j(1-e^{-\lambda iTj})e^{-\lambda iT0}]}$$

Where: λ_i = decay constant of isotope i (1/days)

BP_j^* = steady reactor power (total core power) multiplied by 1/3 to approximate batch power operated in period j (MWt)

T_j^* = duration of operating period j (days)

T_0 = time between the end of operating period j and time of the final reactor shutdown (days)

Σ_j = sum of the batches

Σ_k = the operation of calculating the denominator of the inventory correction for each batch and then summing the batch results prior to division

* For each time period, T_j^*, the variation of steady reactor power, P_j^*, should be limited to ± 20%.

2. Each fission product must be corrected for either 6 half-lives or 3 fuel cycles whichever is shorter. The times are delineated in steps 3 and 4 as the "Power Correction Time".

3. Liquid Sample

Fission Product	Power Correction Time	λ (1/day)	3293 * $(1-e^{-1095\lambda i})$	F_i
I-131	49 days	8.621E-2	3.293E3	
I-133	6 days	7.998E-1	3.293E3	
I-135	2 days	2.517E+0	3.293E3	
Cs-134	3 fuel cycles	9.219E-4	2.089E3	
Cs-137	3 fuel cycles	6.294E-5	2.192E2	

ATTACHMENT 5
Page 3 of 3

4. Gas Sample

Fission Product	Power Correction Time	λ (1/day)	$3293 * (1-e^{-1095\lambda_i})$	F_i
Kr-85m	2 days	3.713E+0	3.293E3	
Kr-85	3 fuel cycles	1.771E-4	5.800E2	
Xe-133	35 days	1.320E-1	3.293E3	
Xe-135	3 days	1.826E+0	3.293E3	

ATTACHMENT 6
Page 1 of 3
NORMALIZED CONCENTRATION OF FISSION PRODUCTS

1. For each fission product in steps 2 and 3 of Attachment 6 perform the following calculation using the applicable data from Attachment 4 and Attachment 5.

$C_w = C_t * F_i$

Where: C_w = the normalized concentration of the fission product (uCi/g for liquids and uCi/cc for gases)

C_t = the decay and pressure/temperature corrected fission product concentration from Attachment 4.

F_i = the inventory correction factor from Attachment 5.

2. Liquid sample - Activity concentrations dispersed equally through reactor water and torus water

Fission Product	C_t	F_i	C_w
I-131			
I-133			
I-135			
Cs-134			
Cs-137			

3. Gas Sample - Activity concentrations dispersed equally through drywell and torus free volumes

Fission Product	C_t	F_i	C_w
Kr-85m			
Kr-85			
Xe-133			
Xe-135			

4. Additional normalizations may be required if plant parameters indicate that the specific activity from a liquid sample represent a sample environment different than the reference environment. Concentration or dilution corrections should be performed and documented in step 7. Reference and typical constants required for the corrections are delineated in step 6.

   Reference mass = the total mass of the reactor water and torus water

   If the actual mass of liquid water does not equal the reference mass a correction factor should be applied.

   $F_{d/c}$ = actual mass (g) / reference mass (g)

5. Additional normalizations may be required if plant parameters indicate that the specific activity from a gas sample represents a sample environment different than the reference environment. Concentrations of dilution corrections should be performed and documented in step 7. Reference and some typical constants required for the corrections are delineated in step 6.

   Reference volume = drywell plus torus free volume

   If the actual volume of gas does not equal the reference volume a correction factor should be applied.

   $F_{d/c}$ = actual mass (cc) / reference mass (cc)

6. Dilution/Concentration Data

Reference liquid mass	3.633E9 g (8.00E6 lbs)
Reactor liquid mass	
At Power	2.93E8 g (6.46E5 lbs)
Hot Standby	3.03E8 g (6.68E5 lbs)
Cold Shutdown	4.09E8 g (9.02E5 lbs)
Torus liquid mass	3.34E9 g (1.33E5 lbs)
Reference gas volume	8.57E9 cc (3.02E5 ft^3)
Torus free volume	3.78E9 cc (1.33E5 ft^3)
Drywell free volume	4.79E9 cc (1.69E5 ft^3)

7. Additional Normalizations

Fission Product	C_w times	$F_{d/c}$ equals	C_{wn}
I-131			
I-133			
I-135			
Cs-134			
Cs-137			
Kr-85m			
Kr-85			
Xe-133			
Xe-135			

ATTACHMENT 7
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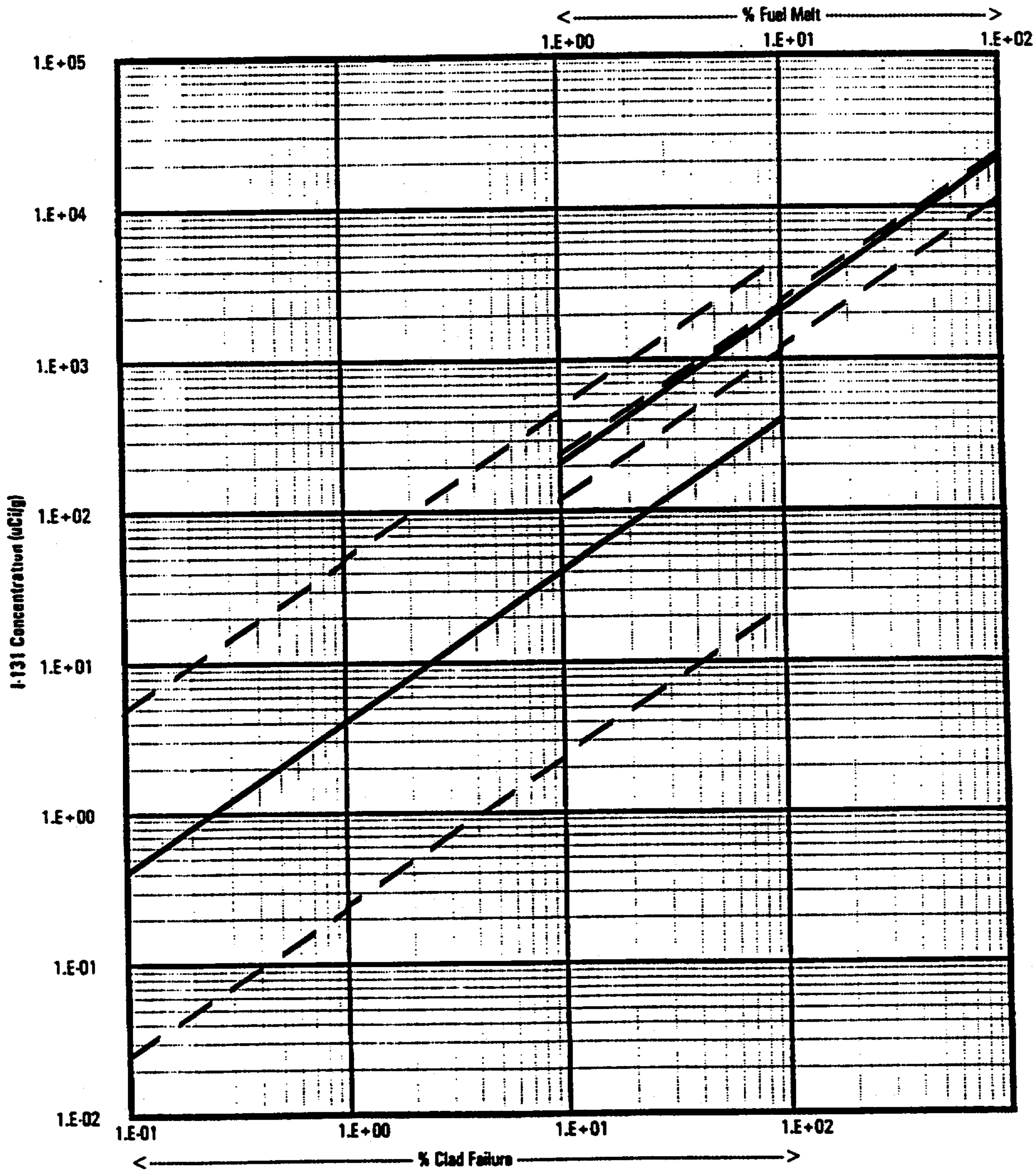
I-131 CONCENTRATION VS. INDICATION OF CORE DAMAGE

I-131 Concentration vs. Indication of Core Damage



ATTACHMENT 8
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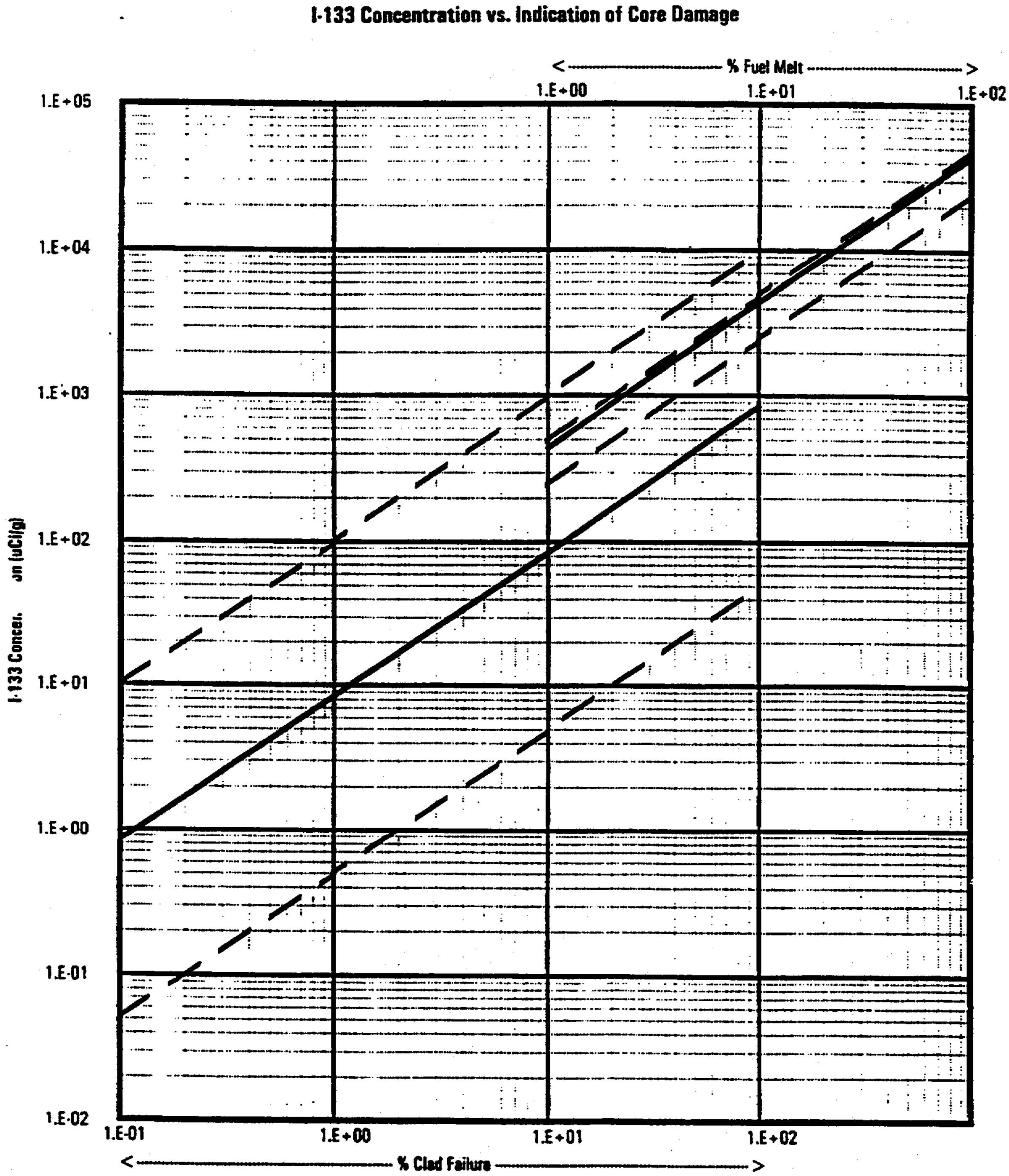
I-133 CONCENTRATION VS. INDICATION OF CORE DAMAGE



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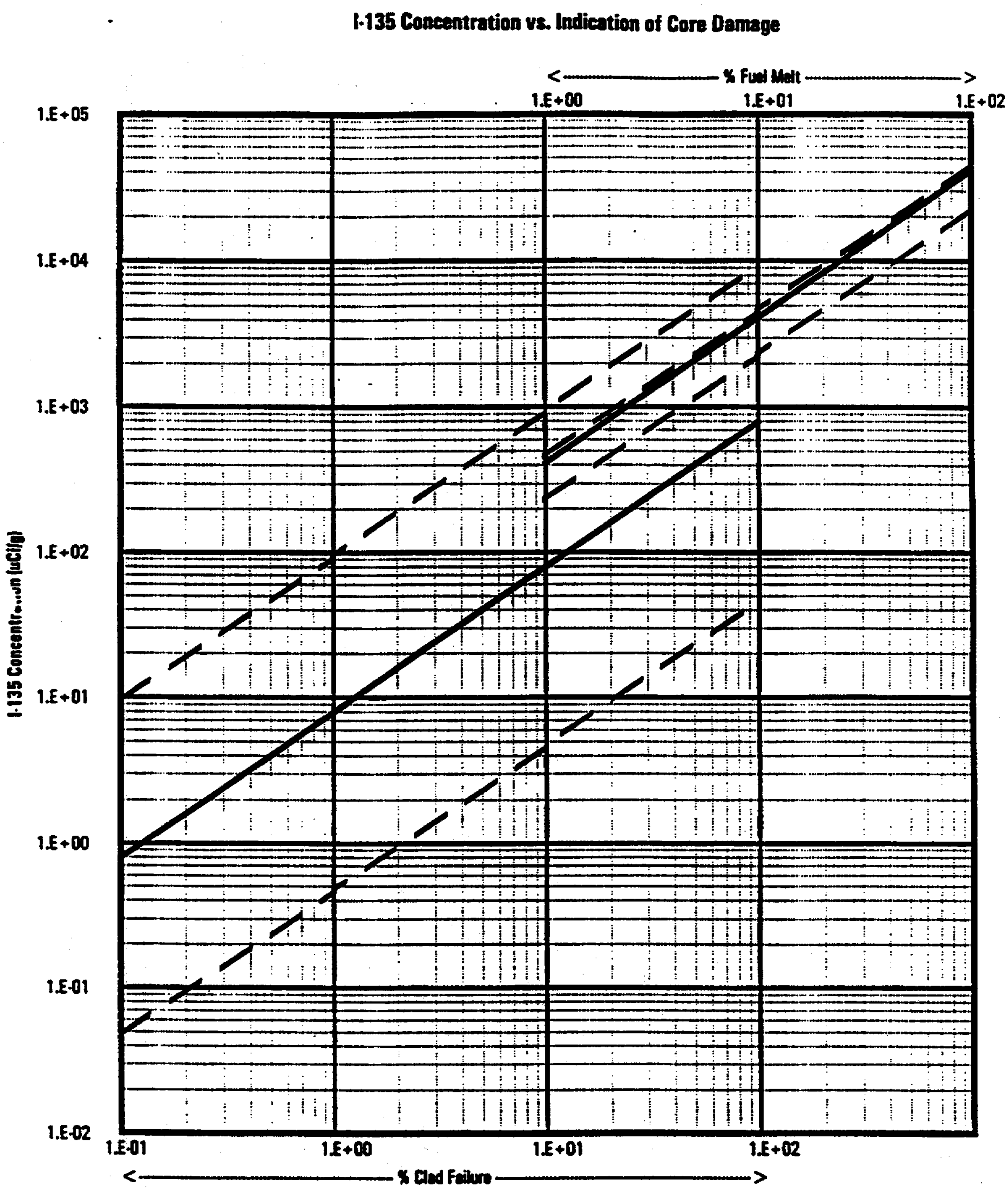
I-135 CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 10
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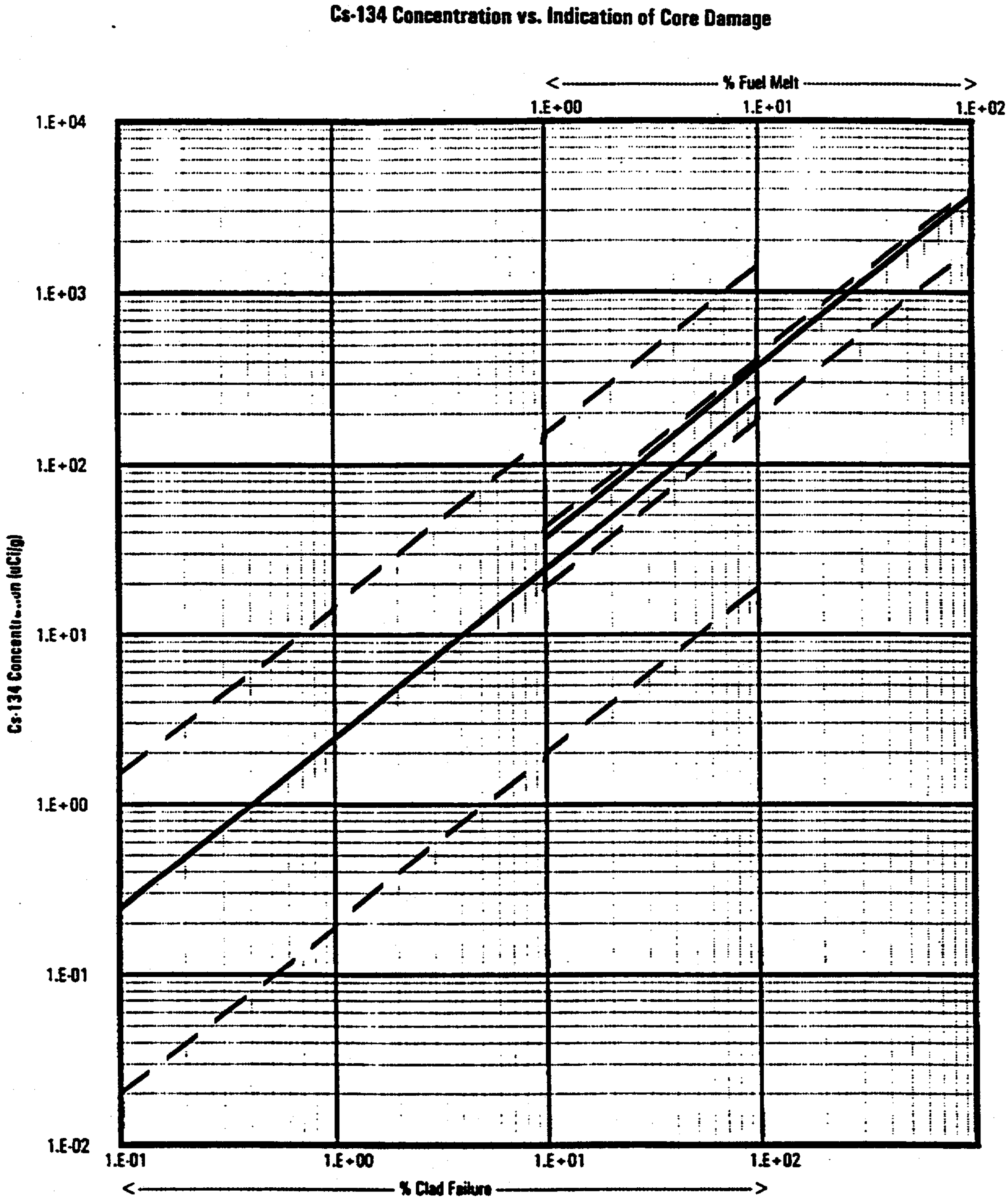
Cs-134 CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 11
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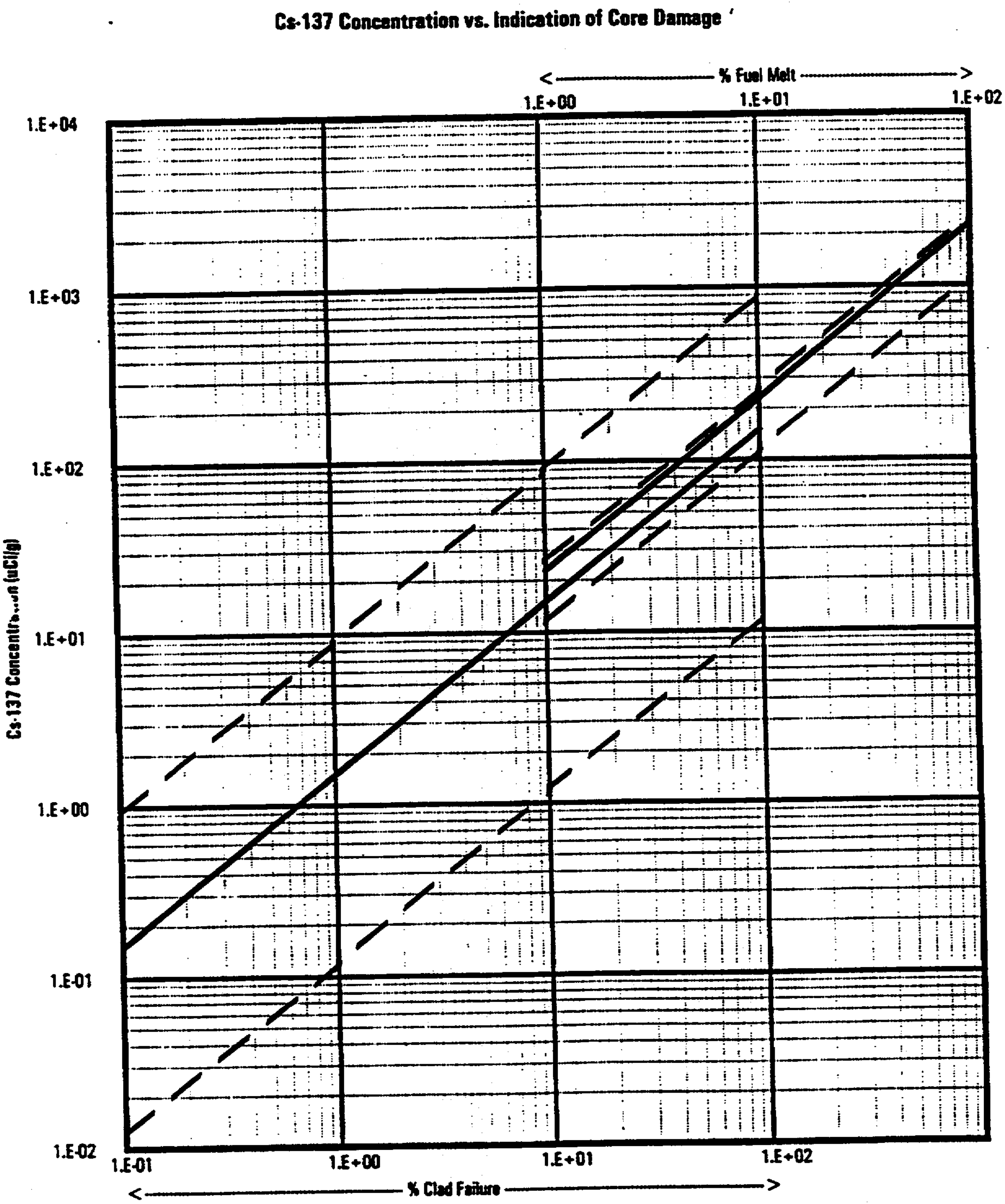
Cs-137 CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 12
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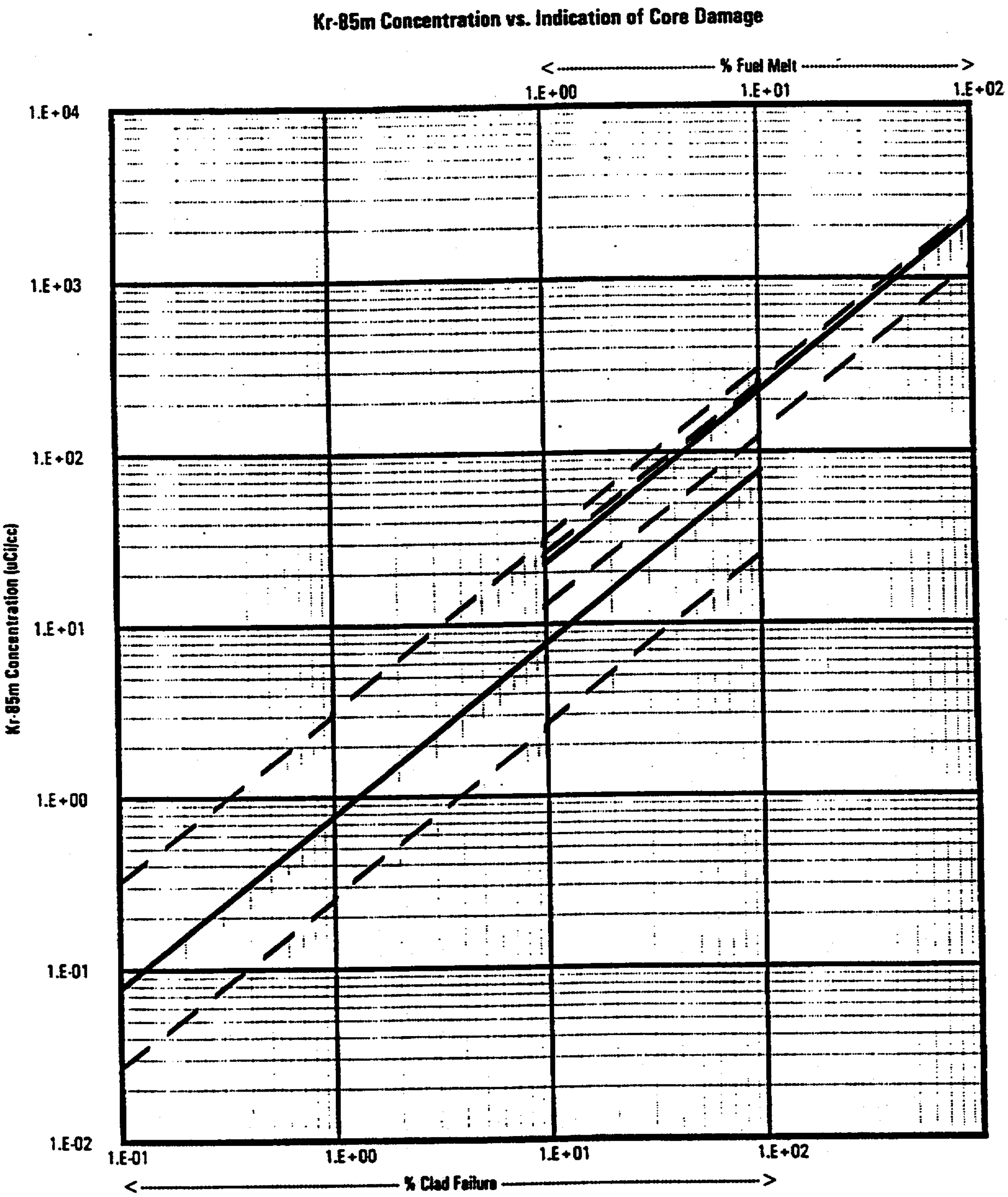
Kr-85m CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 13
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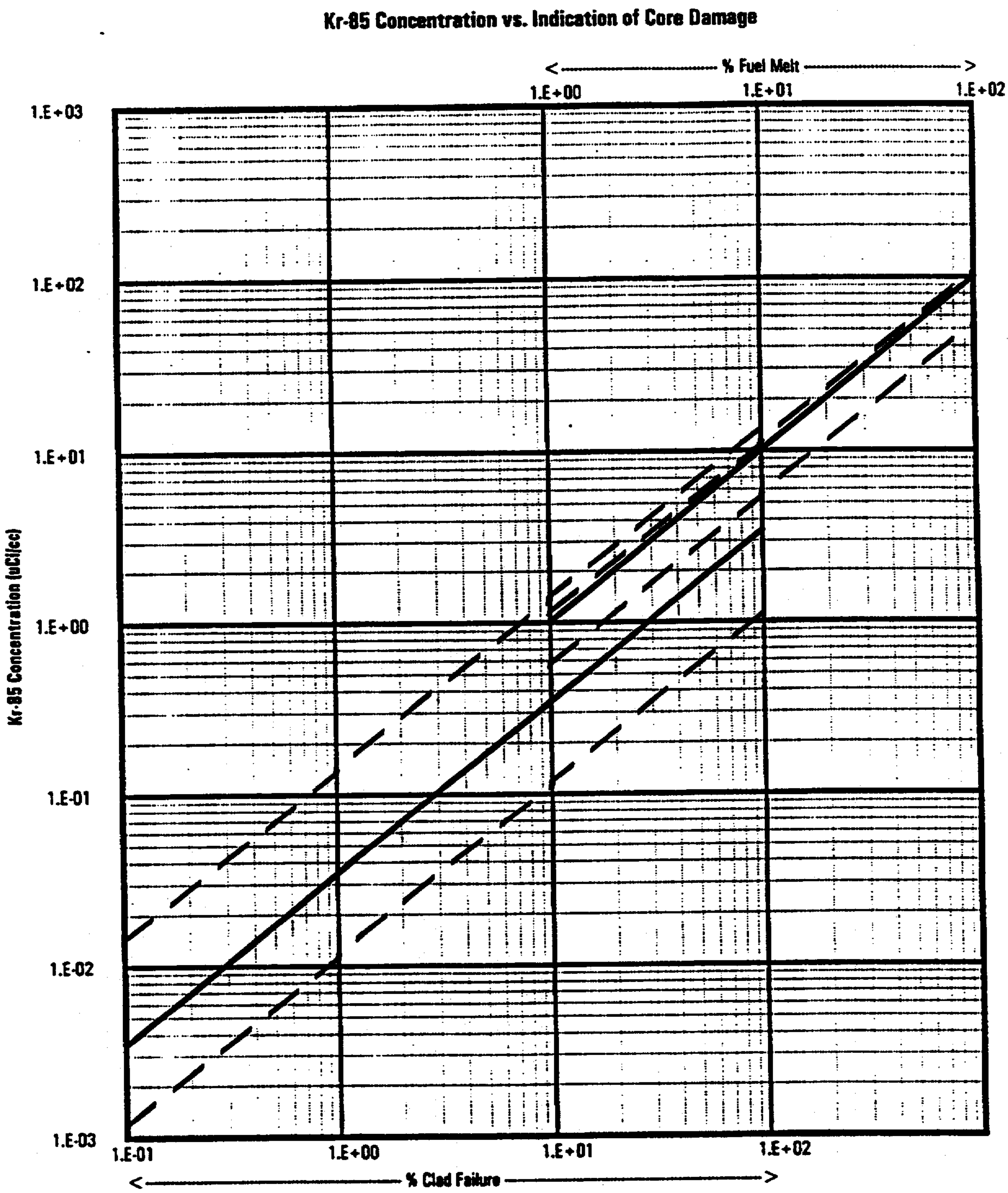
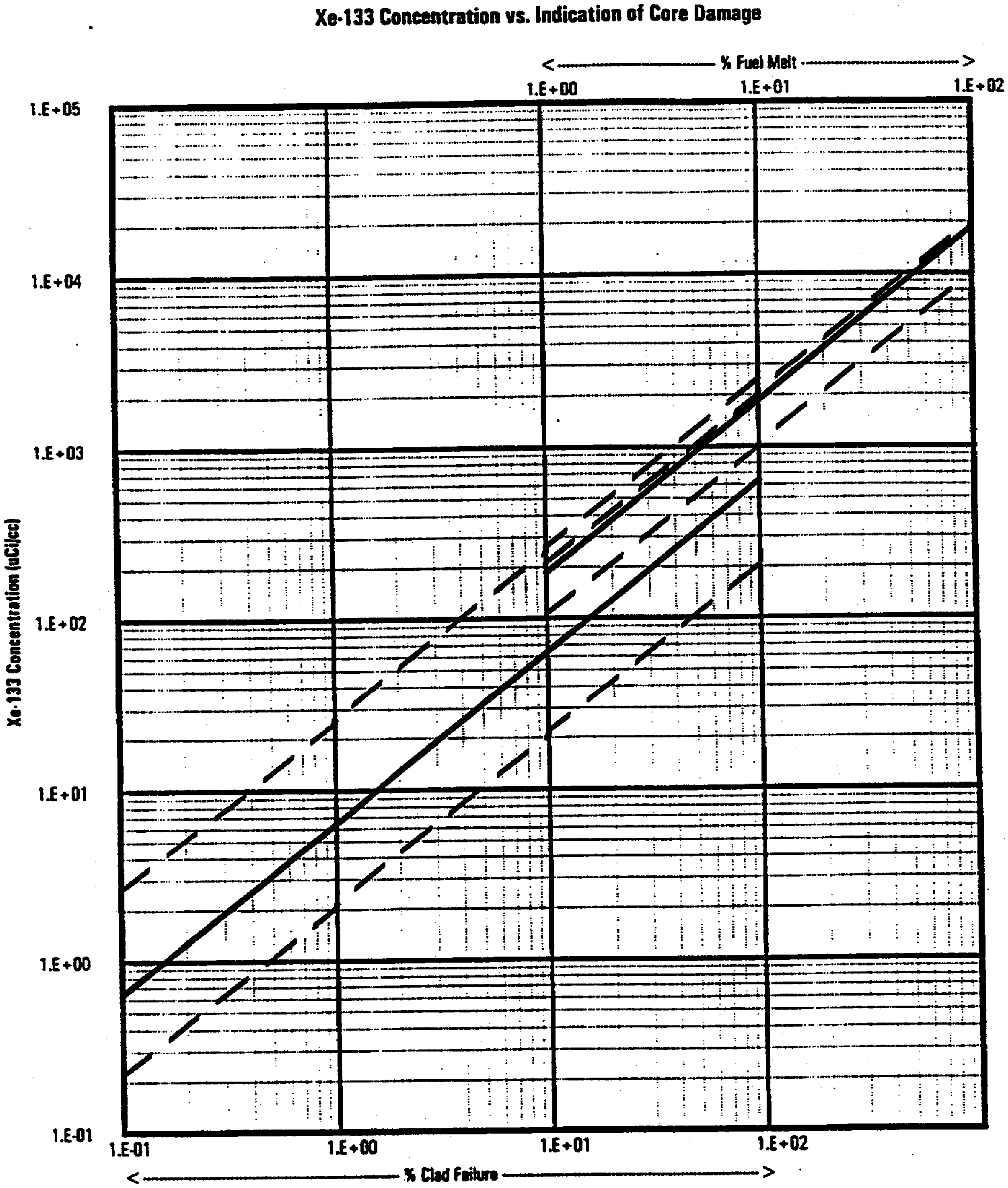
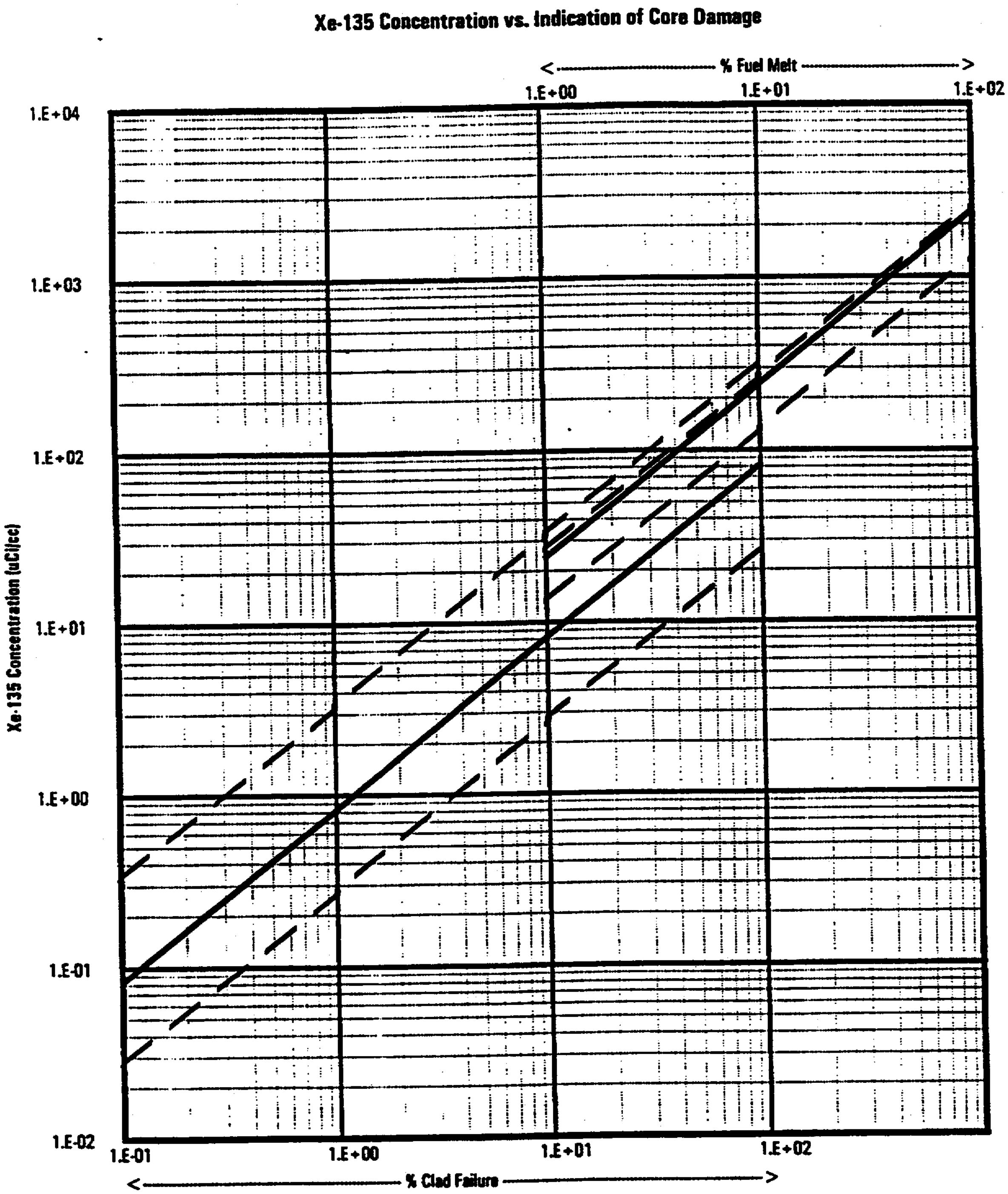
Kr-85m CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 14
Page 1 of 1
Xe-133 CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 15
Page 1 of 1
Xe-135 CONCENTRATION VS. INDICATION OF CORE DAMAGE



ATTACHMENT 16
Page 1 of 2
ISOTOPIC RATIO INDICATION OF RELEASE SOURCE

1. Obtain the decay corrected fission products from Attachment 4 and calculate the ratios as described in step 2.

2. $\frac{\text{Kr-85m}}{\text{Xe-133}}$ = ______________________

   $\frac{\text{Kr-87}}{\text{Xe-133}}$ = ______________________

   $\frac{\text{Kr-88}}{\text{Xe-133}}$ = ______________________

   $\frac{\text{I-132}}{\text{I-131}}$ = ______________________

   $\frac{\text{I-133}}{\text{I-131}}$ = ______________________

   $\frac{\text{I-134}}{\text{I-131}}$ = ______________________

   $\frac{\text{I-135}}{\text{I-131}}$ = ______________________

ATTACHMENT 16
Page 2 of 2

Fission Product Ratio	Ratio in Pellet (indicates fuel melt)	Ratio in Pellet/Clad Gap (indicates clad damage)
Kr-85m / Xe-133	0.122	0.023
Kr-87 / Xe-133	0.233	0.0234
Kr-88 / Xe-133	0.33	0.0495
I-132 / I-131	1.46	0.127
I-133 / I-131	2.09	0.685
I-134 / I-131	2.3	0.155
I-135 / I-131	1.97	0.364

ATTACHMENT 17
Page 1 of 3
CORE DAMAGE ASSESSMENT SUMMARY, DETERMINATION AND RECOMMENDATIONS

Date: ________________ Time: ______________ Summary No.: __________

1. Assessment of amount and type of core damage based on DAPA readings.

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2. Assessment of the % metal-water reaction and corresponding clad failure (determine in conjunction with assessment of adequacy of core cooling if possible).

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3. Assessment of the adequacy of core cooling.

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4. Assessment of release source based on isotopic ratios.

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5. Observations of less volatile fission products.

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6. Core damage estimates based on fission product concentrations from post accident samples as determined utilizing Attachments 7-15.

Fission Product	% Clad Failure	% Fuel Melt
	Liquid Sample	
I-131		
I-133		
I-135		
Cs-134		
Cs-137		
	Gas Sample	
Kr-85m		
Kr-85		
Xe-133		
Xe-135		

NOTES

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ATTACHMENT 17
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7. Summary, Determinations and Recommendations

8. Final Core Damage Estimate

Nuclear Fuels Engineer | Date | Time