TITLE: REACTOR CORE DAMAGE ESTIMATION

AUTHORITY

Function	Signature	Date
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PROCEDURE HISTORY

Revision	Date	Revision	Date	Revision	Date

CONTENTS

1.0 INTRODUCTION

1.1 The purpose of this procedure is to determine the degree of reactor core damage based on water and gas samples taken from the primary system, and drywell radiation levels during accident conditions.

1.2 There are four general classes of fuel damage and three degrees of damage within each of the classes except for the "No Fuel Damage" class.

Class of Fuel Damage	Minor (<10%)	Intermediate (10%-50%)	Major (>50%)
No Fuel Damage	1	1	1
Cladding Failure	2	3	4
Fuel Overheat	5	6	7
Fuel Melt	8	9	10

The objective of this procedure is to narrow down, to the maximum extent possible, those categories which apply to an actual inplant situation.

1.3 In determining the extent of core damage, an initial core damage assessment will be made based on radionuclide measurement.

This initial assessment consists of:

a) Obtaining samples from the Post-Accident Sampling System (PASS).

b) Analyzing the samples for major fission product concentrations by gamma ray spectrometry.

c) Decay correcting samples to the time of reactor shutdown.

d) Normalizing the sample concentrations with reference plant data from a BWR-6/238 with a Mark 3 Containment.

e) Comparing the normalized concentrations to Reference Plant Concentrations Vs. Core Damage Graphs developed by General Electric to estimate the amount of core damage.

This initial core damage assessment will provide one or more candidate categories of possible core damage which will most likely represent the actual in-plant condition.

After the initial assessment is made other parameters should then be evaluated to corroborate and further refine the initial estimate. These parameters should include:

a) Containment hydrogen levels which provide a measure of the extent of metal water reaction which, in turn, can be used to estimate the degree of clad damage.

b) Drywell radiation levels, which measure core damage by an indication of the inventory of airborne fission products (i.e., noble gases, a fraction of the halogens, and smaller fraction of the particulates) released from the fuel to the drywell.

c) Reactor Vessel water level, which 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.

d) Some shorter lived isotope concentrations can be measured from the reactor water and containment gas samples. The ratios of these isotopes can be used to determine the source of the release (fuel or gap).

e) Less volatile fission product concentrations such as isotopes of Sr, Ba, La, and Ru can be measured. If unusually high concentrations in the water samples are found, some degree of fuel melting may be inferred.

The flow diagram in Attachment 1, indicates how the analysis based on radionuclide measurements, and the analysis of other significant parameters relates to the estimation of core damage.

2.0 RESPONSIBILITY

2.1 The Station Emergency Director is responsible for the implementation of this procedure.

2.2 The Supervisor-Emergency Planning is responsible for review of this procedure.

2.3 The Station Emergency Director with advice of his staff will determine sample locations and sample times.

2.4 The Technical Assessment Supervisor is responsible for performing the calculations in this procedure leading to the determination of core damage.

2.5 The Chemistry Department is responsible for obtaining samples from the PASS System and analyzing samples to determine fission product concentrations.

2.6 Radiation Protection is responsible for setting up radiological controls needed to obtain PASS Samples.

3.0 DEFINITIONS

None

4.0 INSTRUCTIONS

4.1 Core Damage Estimate From PASS

4.1.1 Obtain samples, consistent with Appendix H, from the Post Accident Sampling System, per RA-09, POST ACCIDENT SAMPLING. Record the Sample location, clock time, date, Drywell pressure, Containment pressure, Containment temp, Drywell temp, gas sample pressure, and gas sample temp on CORE DAMAGE ESTIMATE FROM PASS DATA SHEET (Attachment 2).

NOTE

Data for this procedure may be obtained from the status boards or computer operator.

4.1.2 Perform gamma spectroscopy per CPS No. 6103.01, GAMMA SPECTROMETER - GELI to determine the I-131 and Cs-137 concentrations in the water samples and Xe-133 and Kr-85 concentrations in the gas samples. Record the isotope concentrations on CORE DAMAGE ESTIMATE FROM PASS DATA SHEET (Attachment 2).

NOTE

If the Gamma Spectroscopy System is setup to give concentrations of isotopes in liquids in units of μCi/ml, convert ml to grams by 1ml=1gm.

NOTE

Measurements of Cs-137 and Kr-85 activities may not be possible until the reactor has been shut down for longer than a few weeks and most of the shorter lived isotopes have decayed.

4.1.3 Correct the measured gaseous activity concentration for temperature and pressure by:

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

Where:

C_{gi} = Containment/Drywell isotopic concentration (μCi/cc)

C_{gi} (Vial) = Sample Vial isotopic Concentration (μCi/cc)

(P_1, T_1) = Sample Vial pressure and temperature on absolute scales (°K, psia)

(P_2, T_2) = Containment/Drywell pressure and temperature on absolute scales (°K, psia)

Record the calculated value for C_{gi} on CORE DAMAGE ESTIMATE FROM PASS DATA SHEET (Attachment 2).

4.1.4 If the fission product concentrations are measured separately for the reactor water and suppression pool water or the drywell gas and the containment gas, the measured concentrations C_{wi} or C_{gi} should be averaged from the separate measurements by:

$\bar{C}_{wi}$ = Average fission product concentration in CPS Coolant (μCi/g)

$$\bar{C}_{wi} = \frac{\text{Isotope Conc. In Rx WTR}\ (2.13 \times 10^{8}) + \text{Isotope Conc. In Supp Pool}\ (4.09 \times 10^{9})}{(4.3 \times 10^{9})}$$

$\bar{C}_{gi}$ = Average fission product concentration in Containment gas (Ci/cc)

$$\bar{C}_{gi} = \frac{\text{Isotope Conc In Drywell}\ (6.98 \times 10^{9}) + \text{Isotope Conc. In CNMT}\ (3.7 \times 10^{10})}{(4.398 \times 10^{10})}$$

Record the calculated values $\bar{C}_{wi}$ and $\bar{C}_{gi}$ on CORE DAMAGE ESTIMATE FROM PASS DATA SHEET, (Attachment 2.)

4.1.5 Calculate the fission product inventory correction factor F_{Ii} for I-131, Cs-137, Xe-133, and Kr-85 by:

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

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

Where:

P_j = Steady reactor power operated in period j (MW_t)*

T_j = Duration of operating period j (day)*

T_j° = Time between the end of operating period j and time of last reactor shutdown (day)

λ_i = Decay Constant for nuclide i (Day^{-1})

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

Record the calculated fission product inventory correction factors on CORE DAMAGE ESTIMATE FROM PASS DATA SHEET, Attachment 2.

NOTE

See Appendix J for a sample calculation

4.1.6 Calculate the normalized concentration C_{Wi}^{Ref} for I-131 and Cs-137, and C_{gi}^{Ref} for Xe-133 and Kr-85 by:

$$C_{Wi}^{Ref} = \bar{C}_{Wi}\, e^{-\lambda_i t} \times F_{Ii} \times (1.10)$$

$$C_{gi}^{Ref} = \bar{C}_{gi}\, e^{-\lambda_i t} \times F_{Ii} \times (1.10)$$

Where:

C_{Wi}^{Ref} = normalized concentration of isotope i for reference plant coolant (μCi/g)

C_{gi}^{Ref} = normalized concentration of isotope i for reference plant containment gas (μCi/cc)

$\bar{C}_{Wi}$ = average concentration of isotope i in CPS coolant at time, t (μCi/g)

$\bar{C}_{gi}$ = average concentration of isotope i in CPS containment gas at time, t (μCi/cc)

λ_i = decay constant of isotope i (day^{-1})

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

F_{Ii} = inventory correction factor for isotope i

Record calculated normalized concentrations for reference plant coolant and containment gas on CORE DAMAGE ESTIMATE FROM PASS DATA SHEET (Attachment 2).

4.1.7 If the normalized concentrations, C_{Wi}^{Ref} or C_{gi}^{Ref}, obtained in section 4.1.6 are higher than the Upper Limit concentrations shown in Appendix B the extent of fuel or cladding damage can be determined directly from Appendix D through G. Record the estimated fuel/clad damage on CORE DAMAGE ESTIMATE FROM PASS DATA SHEET (Attachment 2).

4.2 Hydrogen Analysis

4.2.1 Obtain a containment hydrogen and oxygen gas concentration reading from the Containment H_2/O_2 Atmospheric Monitoring (CAM) system (%). The reading to be selected should be based on engineering judgement. Record readings on HYDROGEN ANALYSIS DATA SHEET (Attachment 3).

NOTE

The calculation for percent metal-water reaction is based on perfect hydrogen mixing in the containment. Gas concentration readings should be taken in the drywell and containment to verify that this is the case.

4.2.2 Calculate the Decimal Equivalent Metal-Water Reaction (MWR) Per:

$$MWR = 2.422 \left[\frac{\chi H_2 - 2\chi O_2}{1 - \chi H_2 - \chi O_2}\right] + 1.284$$

Where: χH_2 = Percent hydrogen concentration (decimal equivalent)

χO_2 = Percent oxygen concentration (decimal equivalent)

Record calculated value for MWR on HYDROGEN ANALYSIS DATA SHEET (Attachment 3).

4.3 Drywell Radiation Analysis

4.3.1 Obtain Drywell Atmosphere Monitoring readings, [R] in R/hr from 1RIX-CM059(1H13-P638) and 1RIX-CM060(1H13-P639). Record readings on DRYWELL RADIATION ANALYSIS DATA SHEET (Attachment 4).

NOTE

The two drywell Monitor Readings should be averaged for use in Appendix K calculations. If the two readings are in disagreement by more than an order of magnitude a determination should be made as to the validity of the data.

4.3.2 Determine elapsed time [T] in hours from plant shutdown to the Drywell radiation monitor reading and record on DRYWELL RADIATION ANALYSIS DATA SHEET (Attachment 4).

4.3.3 From Appendix K, determine the fuel inventory release for the reference plant [I ref] %, and record on DRYWELL RADIATION ANALYSIS DATA SHEET, (Attachment 4).

4.3.4 Determine the inventory release [I] to the CPS drywell using the following formula:

$$I = 1.3 \times I_{ref}$$

Record inventory release to the drywell on DRYWELL RADIATION ANALYSIS SHEET (Attachment 4).

4.4 Reactor Core Uncovery Time

NOTE

The graph of Maximum Acceptable Core Uncovery Time Vs. Time After Reactor Shutdown in Appendix (L) was based on the time required for a completely uncovered core to heat up from equilibrium at 545°F to a peak clad temperature of 2,200°F with no spray or steam cooling. If only partial core uncovery or spray cooling occurred, longer maximum acceptable core uncovery times are likely.

4.4.1 From the control room reactor water level instrumentation determine the length of time the reactor core was completely uncovered. Record this data on REACTOR CORE UNCOVERY TIME DATA SHEET (Attachment 5).

4.4.2 Compare the actual core uncovery time determined in step 4.4.1 to the maximum acceptable core uncovery time obtained from Appendix (L) to determine if core damage is likely. Record the maximum acceptable core uncovery time from Appendix (L) on REACTOR CORE UNCOVERY TIME DATA SHEET (Attachment 5).

4.5 Identification of Release Source by Determination of Fission Product Ratios.

4.5.1 From samples obtained from PASS, determine the concentrations of the following short-lived isotopes by gamma ray spectroscopy per CPS No. 6103.01, GAMMA SPECTROMETER - GELI.

Liquid samples (μCi/g)	Gas sample(μCi/cc)
I-131	Kr-85m
I-132	Kr-87
I-133	Kr-88
I-134	Xe-133
I-135	

Record isotopic concentrations on FISSION PRODUCT RATIOS DATA SHEET (Attachment 6).

4.5.2 Correct the measured fission products to the time of reactor shutdown by

$$C_{i,o} = C_{i,t}\, e^{\lambda_i t}$$

Where:

$C_{i,o}$ = concentration of isotope i at shutdown. (μCi/g) or (μCi/cc)

$C_{i,t}$ = measured concentration of isotope i at time t. (μCi/g) or (μCi/cc)

λ_i = decay constant of isotope i (day^{-1}).

t = time between reactor shutdown and sample analysis (day).

Record the Corrected fission product concentrations on FISSION PRODUCT RATIO DATA SHEET (Attachment 6).

4.5.3 Calculate the isotopic ratios by:

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

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

Record the fission product ratios on FISSION PRODUCT RATIO DATA SHEET (Attachment 6).

4.5.4 Determine the release source by comparing the isotopic ratios from step 4.5.3 to the ratios supplied in Appendix C. Record the release source determined by each ratio on FISSION PRODUCT RATIO DATA SHEET (Attachment 6).

NOTE

Generally, lower fission product activity ratios are found in the fuel gap, so lower fission product ratios measured in CPS coolant or containment atmosphere is indicative of fuel cladding failure. Higher fission product activity ratios are found in the core fuel, and higher fission product activity ratios are indicative of fuel melt.

4.6 Analysis for Ba, Sr, La, Ru.

4.6.1 From samples obtained from PASS determine the concentrations of the following short-lived isotopes by gamma ray spectroscopy:

Sr-91 (μCi/g)
Sr-92 (μCi/g)
Ba-140 (μCi/g)
La-140 (μCi/g)
Ru-103 (μCi/g)

Record the isotope concentrations on ANALYSIS FOR Ba, Sr, La and Ru DATA SHEET (Attachment 7).

4.6.2 Compare the isotopic concentrations obtained in step 4.6.1. to the baseline concentration data maintained by the Chemistry Department. If unusually high concentrations of Sr, Ba, La and Ru are found in the water samples (i.e., greater than 100% above baseline), some degree of fuel melting may be inferred.

5.0 REFERENCES

1. NEDO-22215 Procedures For The Determination of the Extent of Core Damage Under Accident Conditions. Dated: August 1982

2. NEDO-22215 Attachment 8D, Procedures for Estimating Core Damage Based on Plant Parameters Other than Post-Accident Sampling System Measurements.

3. BWROG Emergency Procedures Guidelines, Rev. 2 Appendix C: Calculational Procedures, C23.0 Maximum Acceptable Core Uncovery Time.

4. RA-09, POST ACCIDENT SAMPLING

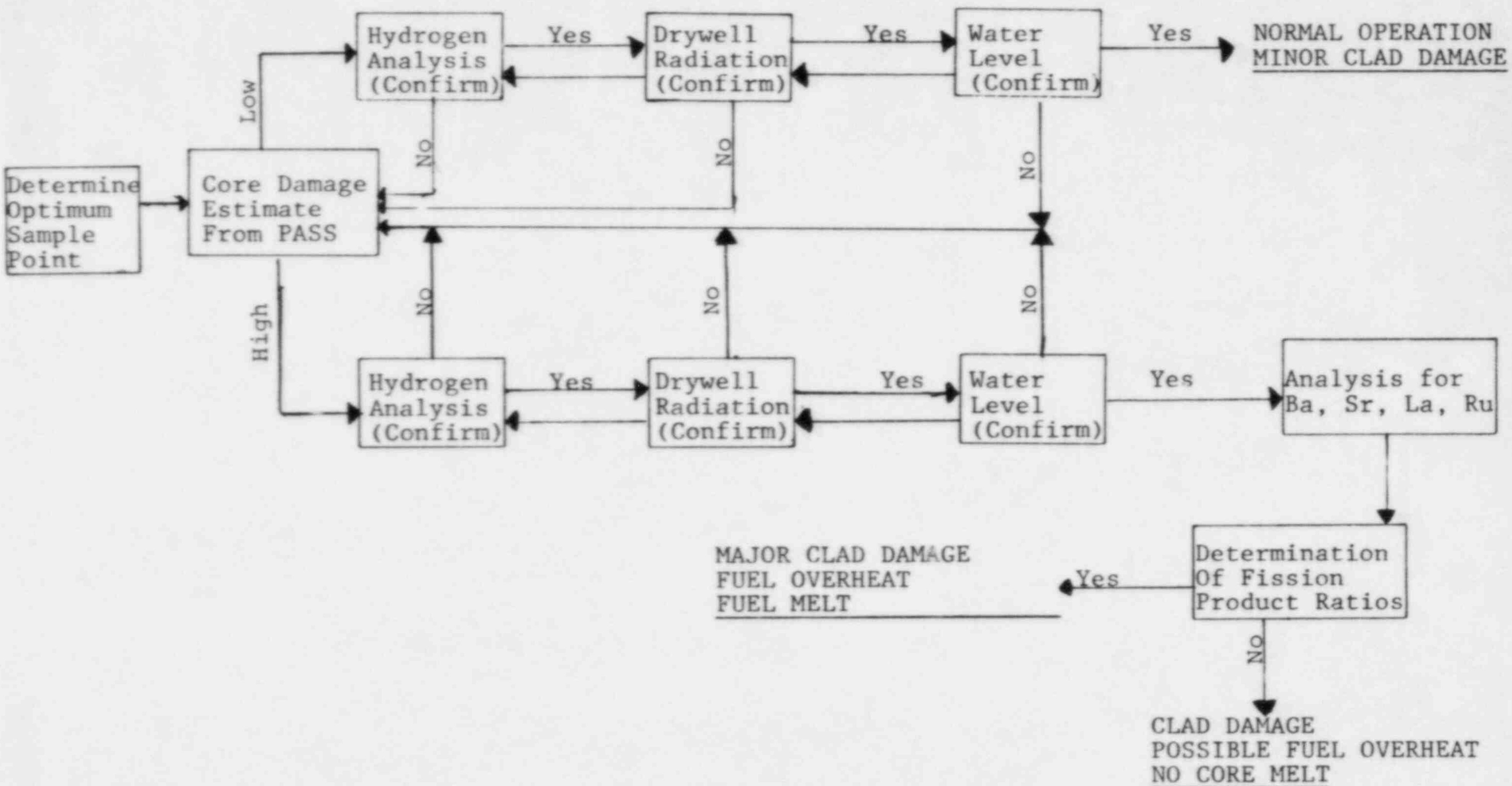
5. CPS No. 6103.01, GAMMA SPECTROMETER-GELI

6. FSAR APPENDIX, TABLE D-1, RADIOACTIVE SOURCE ASSUMPTIONS, (AMENDMENT 14).

6.0 ATTACHMENTS

1. SEQUENCE OF ANALYSIS FOR ESTIMATION OF CORE DAMAGE
2. CORE DAMAGE ESTIMATE FROM PASS DATA SHEET
3. HYDROGEN ANALYSIS DATA SHEET
4. DRYWELL RADIATION ANALYSIS DATA SHEET
5. REACTOR CORE UNCOVERY TIME DATA SHEET
6. FISSION PRODUCT RATIO DATA SHEET
7. ANALYSIS FOR Ba, Sr, La, and Ru

SEQUENCE OF ANALYSIS FOR ESTIMATION OF CORE DAMAGE



CORE DAMAGE ESTIMATE FROM PASS DATA SHEET

Sample Location

	Check Appropriate Location	Sample Time	Analysis Time	(Remarks)
Jet Pump	___	________	________	________
RWCU	___	________	________	________
Supp. Pool Liquid	___	________	________	________
Containment Atmosphere	___	________	________	________
RHR	___	________	________	________
Drywell	___	________	________	________

Drywell Pressure ____________________ (psia)

Drywell Temp. ____________________ (°K)

Containment Pressure ____________________ (psia)

Containment Temp ____________________ (°K)

Containment Atmos Sample Vial Pressure ____________________ (psia)

Containment Atmos Sample Vial Temp ____________________ (°K)

Drywell Sample Vial Pressure ____________________ (psia)

Drywell Sample Vial Temp ____________________ (°K)

Isotopic Concentration of Sample

Sample Location	Isotope	Concentration	Isotope	Concentration
Jet Pump	I-131	________ (μCi/g)	Cs-137	________ (μCi/g)
RWCU	I-131	________ (μCi/g)	Cs-137	________ (μCi/g)
Supp. Pool Liquid	I-131	________ (μCi/g)	Cs-137	________ (μCi/g)
Containment Atmos	Xe-133	________ (μCi/cc)	Kr-85	________ (μCi/cc)
RHR	I-131	________ (μCi/g)	Cs-137	________ (μCi/g)
Drywell	Xe-133	________ (μCi/cc)	Kr-85	________ (μCi/cc)

Temperature and Pressure Corrected Gas Sample Concentration

	(Xe-133)	(Kr-85)
C_{gi}: Containment Atmos Sample	________ (μCi/cc)	________ (μCi/cc)
C_{gi}: Drywell Atmos Sample	________ (μCi/cc)	________ (μCi/cc)

Average Fission Product Concentrations

$\bar{C}_{wI-131}$: (Average I-131 Concentration in CPS Coolant)= ________ (μCi/g)

$\bar{C}_{wCs-137}$:(Average Cs-137 Concentration in CPS Coolant)= ________ (μCi/g)

$\bar{C}_{gXe-133}$:(Average Xe-133 Concentration in CNMT gas)= ________ (μCi/cc)

$\bar{C}_{gKr-85}$: (Average Kr-85 Concentration in CNMT gas)= ________ (μCi/cc)

Fission Product Correction Factor

F_{II-131}: (I-131 Fission product Correction Factor) = ________

$F_{ICs-137}$: (CS-137 Fission product Correction Factor) = ________

$F_{IXe-133}$: (Xe-133 Fission product Correction Factor) = ________

F_{IKr-85}: (Kr-85 Fission product Correction Factor) = ________

Normalized Fission Product Concentrations for Reference Plant

C^{Ref}_{wI-131}: (I-131 Conc. in reference plant coolant) = ________ (μCi/g)

$C^{Ref}_{wCs-137}$: (Cs-137 Conc. in reference plant coolant) = ________ (μCi/g)

$C^{Ref}_{wXe-133}$: (XE-133 Conc. in reference plant CNMT gas) = ________ (μCi/cc)

C^{Ref}_{wKr-85}: (Kr-85 Conc. in reference plant CNMT gas) = ________ (μCi/cc)

Estimated Fuel/Clad Damage

	Cladding Failure	Fuel Meltdown
Appendix D estimate	________ %	________ %
Appendix E estimate	________ %	________ %
Appendix F estimate	________ %	________ %
Appendix G estimate	________ %	________ %

Remarks

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____

____

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____

Calculations Performed By: ____________________

Time/Date Completed: ____________________

HYDROGEN ANALYSIS DATA SHEET

Containment Hydrogen and Oxygen Gas Concentration

	Containment	Drywell
O_2 Conc. =	________________%	________________%
H_2 Conc. =	________________%	________________%

Metal - Water Reaction

MWR = ________________

Remarks

____
____
____
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____
____
____
____

Performed By: ______________________________

Time/Date: ______________________________

DRYWELL RADIATION ANALYSIS DATA SHEET

Drywell Radiation Monitor Detector ID#	Radiation Reading (R/hr)
1 RIX-CM059(1H13-P638)	______
1 RIX-CM060(1H13-P639)	______

Elapsed time from plant Shutdown to Drywell radiation monitor reading ______________ (hours)

Inventory release for reference Plant determined from App. K ______________ (%)

Calculated inventory release to CPS Drywell ______________ (%)

Remarks

____

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Performed By: ______________________

Time/Date: ______________________

REACTOR CORE UNCOVERY TIME DATA SHEET

Control Room Instrument ID#	Reactor Core Uncovery Time
____________	____________
____________	____________
____________	____________

Maximum Acceptable Core Uncovery Time from App. (L). ____________________

Remarks

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Performed By: ____________________________

Time/Date: ____________________________

FISSION PRODUCT RATIO DATA SHEET

Isotopic Concentrations

Liquid Sample (Isotope)	Concentration (μCi/g)	Gas Sample (Isotope)	Concentration (μCi/cc)
I-131	________	Kr-85m	________
I-132	________	Kr-87	________
I-133	________	Kr-88	________
I-134	________	Xe-133	________
I-135	________		

Corrected Fission Product Concentrations

Liquid Sample (Isotope)	Concentration (μCi/g)	Gas Sample (Isotope)	Concentration (μCi/cc)
I-131	________	Kr-85m	________
I-132	________	Kr-87	________
I-133	________	Kr-88	________
I-134	________	Xe-133	________
I-135	________		

Isotopic Ratios

Noble Gas Ratios

$\frac{Kr\text{-}85m}{Xe\text{-}133}$ = ________

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

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

Iodine Ratios

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

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

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

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

Release Source

Ratio	Release Source (Core Inventory/Fuel Gap)	Ratio	Release Source (Core Inventory/Fuel Gap)
$\frac{Kr\text{-}85m}{Xe\text{-}133}$ =	______	$\frac{I\text{-}132}{I\text{-}131}$ =	______
$\frac{Kr\text{-}87}{Xe\text{-}133}$ =	______	$\frac{I\text{-}133}{I\text{-}131}$ =	______
$\frac{Kr\text{-}88}{Xe\text{-}133}$ =	______	$\frac{I\text{-}134}{I\text{-}131}$ =	______
		$\frac{I\text{-}135}{I\text{-}131}$ =	______

Performed By: ______________________

Time/Date: ______________________

ANALYSIS FOR Ba, Sr, La, and Ru, DATA SHEET

Isotope	Measured Concentration (μCi/g)
Sr-91	________
Sr-92	________
Ba-140	________
La-140	________
Ru-103	________

Performed By: ____________________

Time/Date: ____________________

TITLE: REACTOR CORE DAMAGE ESTIMATION

CORE INVENTORY OF MAJOR FISSION PRODUCTS IN A REFERENCE PLANT OPERATED AT 3651 MW_t 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(0.753)
	Kr-85	10.72y	1.1	514(0.0044)
	Kr-87	76.3m	47.1	403(0.495)
	Kr-88	2.84h	66.8	196(0.26),1530(0.109)
	Xe-133	5.25d	202.0	81(0.365)
	Xe-135	9.11h	26.1	250(0.899)
Halogens	I-131	8.04d	96.0	364(0.812)
	I-132	2.3h	140	668(0.99,773(0.762)
	I-133	20.8h	201	530(0.86)
	I-134	52.6m	221	847(0.954),884(0.653)
	I-135	6.61h	189	1132(0.225),1260(0.286)
Alkali Metals	Cs-134	2.06y	19.6	605(0.98),796(0.85)
	Cs-137	30.17y	12.1	662(0.85)
	Cs-138	32.2m	178.0	463(0.307),1436(0.76)
Tellurium Group	Te-132	78.2h	138	228(0.88)
Noble Metals	Mo-99	66.02h	183	740(0.128)
	Ru-103	39.4d	155	497(0.89)
Alkaline Earths	Sr-91	9.5h	115	750(0.23),1024(0.325)
	Sr-92	2.71h	123	1388(0.9)
	Ba-140	12.8d	173	537(0.254)
Rare Earths	Y-92	3.54h	124	934(0.139)
	La-140	40.2h	184	487(0.455),1597(0.955)
	Ce-141	32.5d	161	145(0.48)
	Ce-144	284.3d	129	134(0.108)
Refractories	Zr-95	64.0d	161	724(0.437,757(0.553)
	Zr-97	16.9h	166	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.

TITLE: REACTOR CORE DAMAGE ESTIMATION

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

	Reactor Water, (μCi/g)		Drywell Gas (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	---	---	$1x10^{-4}$[a]	$1x10^{-5}$[b]
Kr-85	---	---	$4x10^{-5}$[a]	$4x10^{-6}$[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.

TITLE: REACTOR CORE DAMAGE ESTIMATION

RATIOS OF ISOTOPES IN CORE INVENTORY AND FUEL GAP

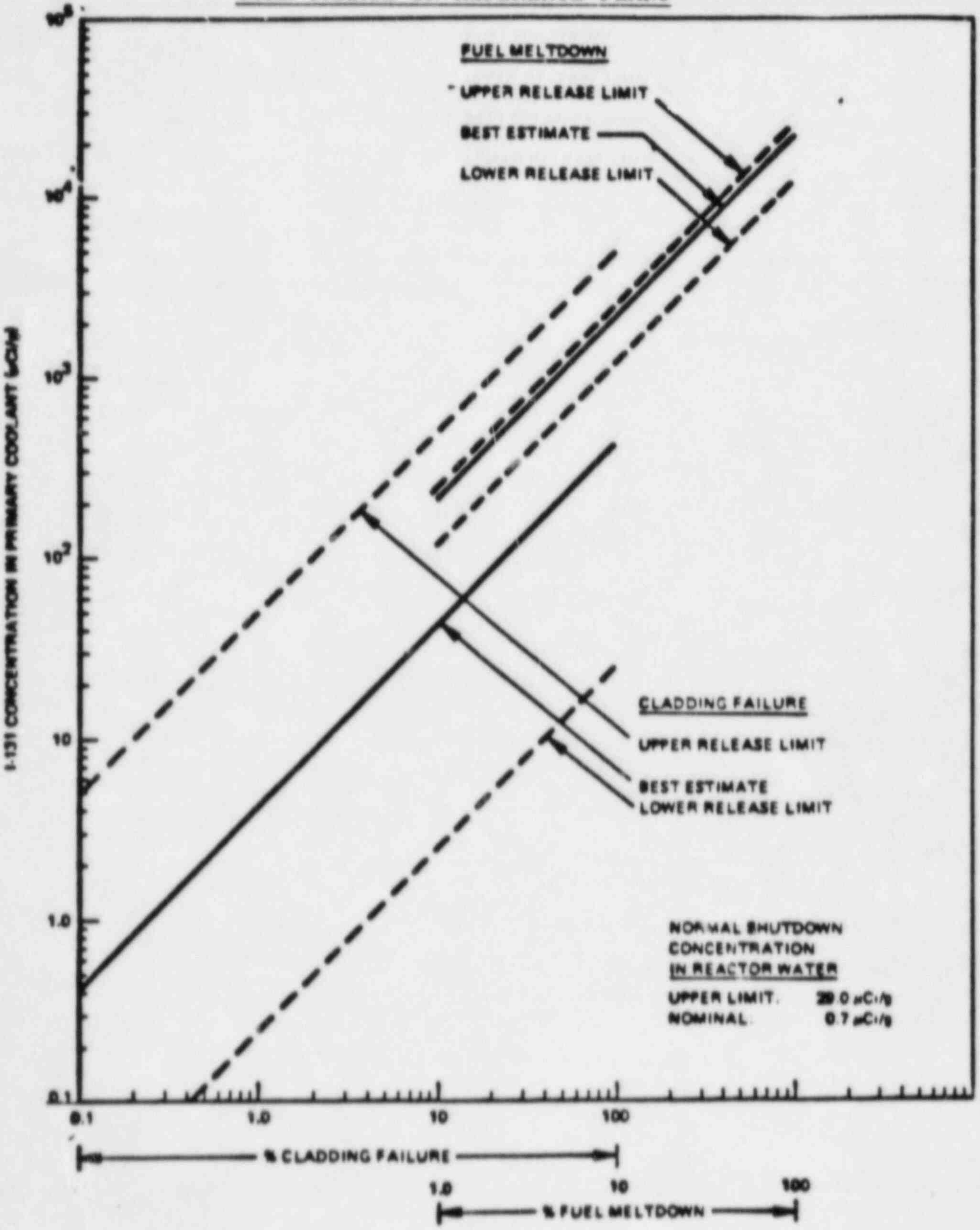
Isotope	Half-Life	Activity Ratio in Core Inventory	Activity Ratio in Fuel Gap
Kr-87	76.3m	0.233	0.0234
Kr-88	2.84h	0.33	0.0495
Kr-85m	4.48h	0.122	0.023
Xe-133	5.25d	1.0	1.0
I-134	52.6m	2.3	0.155
I-132	2.3h	1.46	0.127
I-135	6.61h	1.97	0.364
I-133	20.8h	2.09	0.685
I-131	8.04d	1.0	1.0

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

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

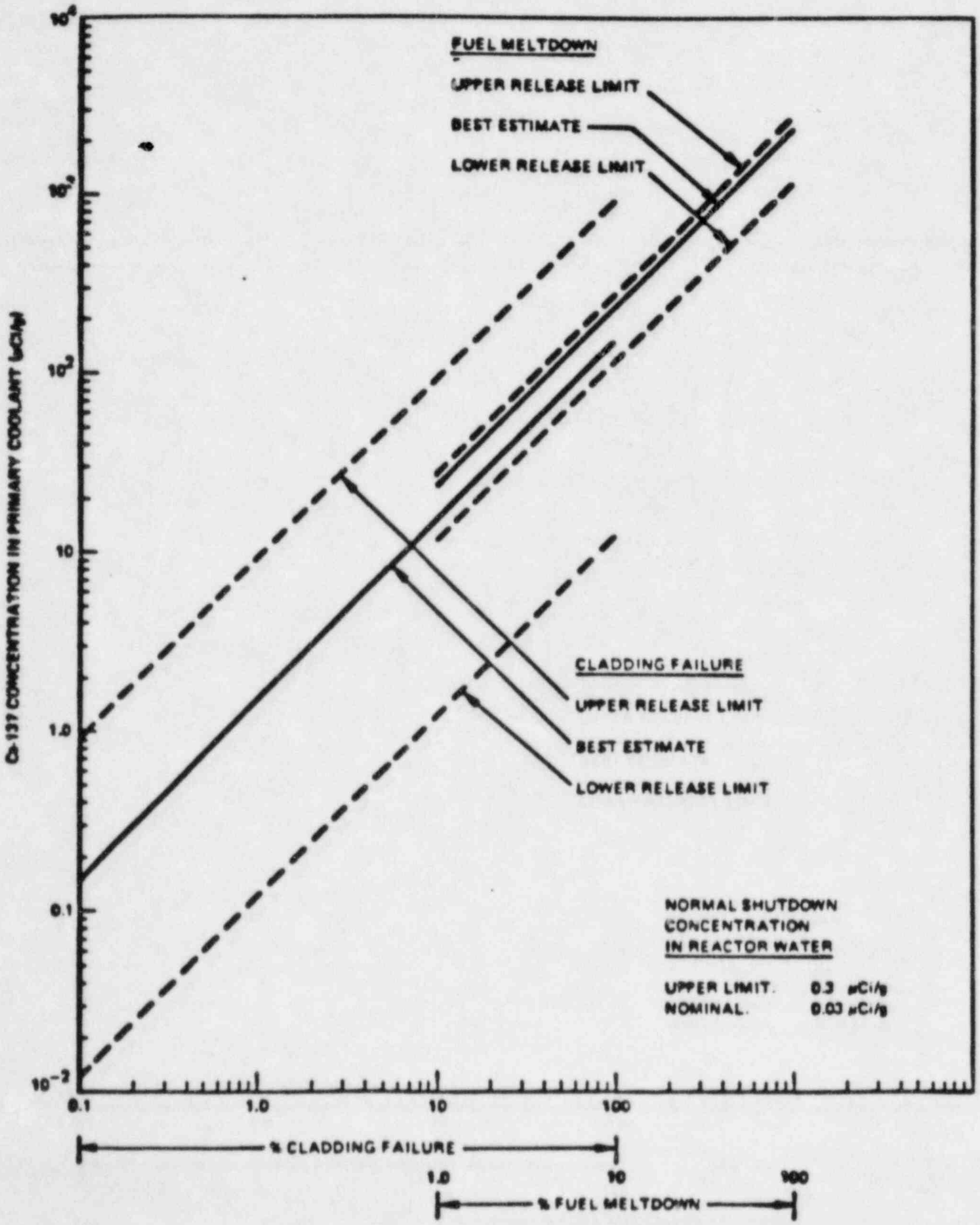
TITLE: REACTOR CORE DAMAGE ESTIMATION

RELATIONSHIP BETWEEN I-131 CONCENTRATION IN THE PRIMARY COOLANT (REACTOR WATER + POOL WATER) AND THE EXTENT OF CORE DAMAGE IN REFERENCE PLANT



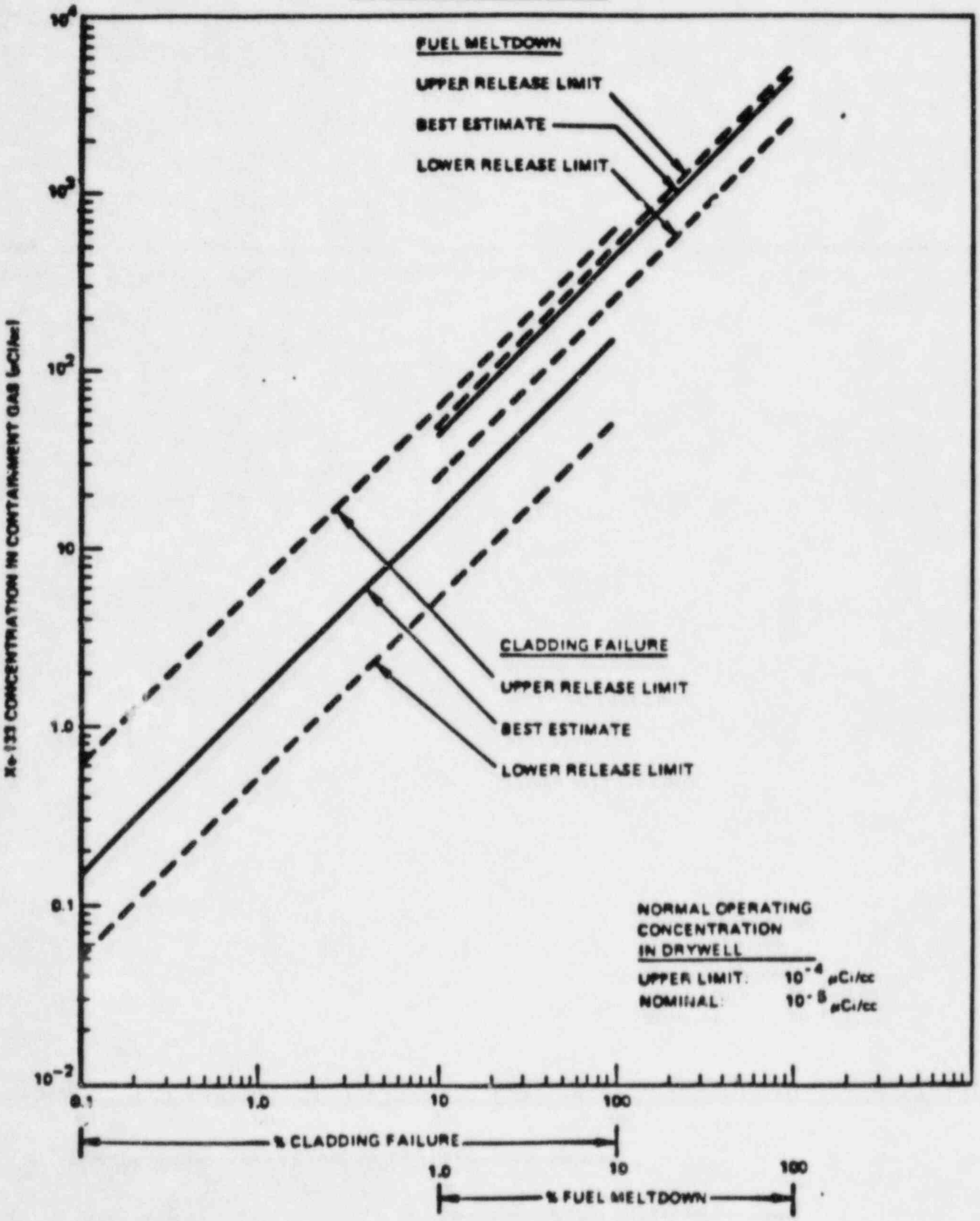
TITLE: REACTOR CORE DAMAGE ESTIMATION

RELATIONSHIP BETWEEN Cs-137 CONCENTRATION IN THE PRIMARY COOLANT (REACTOR WATER + POOL WATER) AND THE EXTENT OF CORE DAMAGE IN REFERENCE PLANT



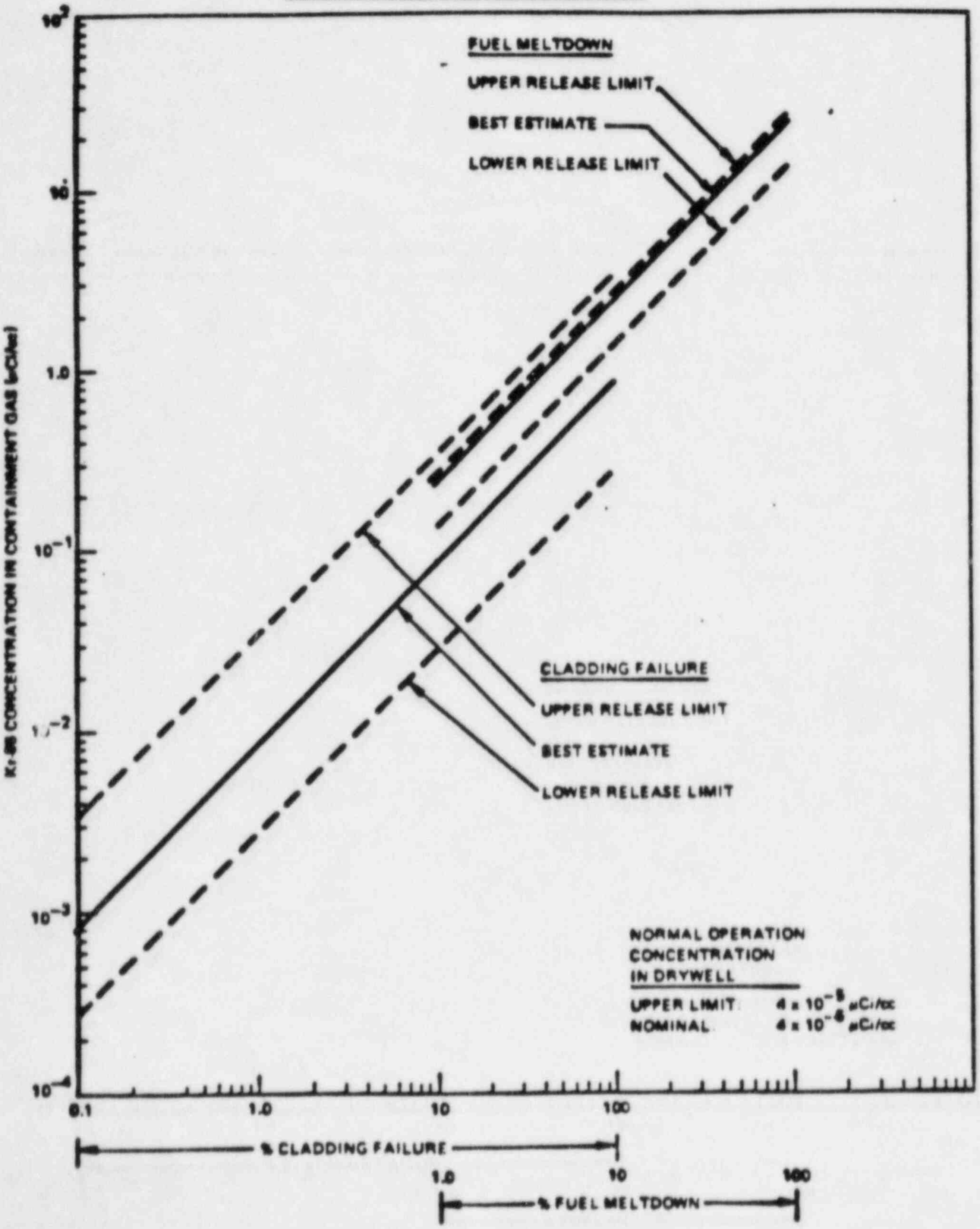
TITLE: REACTOR CORE DAMAGE ESTIMATION

RELATIONSHIP BETWEEN Xe-133 CONCENTRATION IN THE CONTAINMENT GAS (DRYWELL + PRIMARY CONTAINMENT) AND THE EXTENT OF CORE DAMAGE IN REFERENCE PLANT



TITLE: REACTOR CORE DAMAGE ESTIMATION

RELATIONSHIP BETWEEN Kr-85 CONCENTRATION IN THE CONTAINMENT GAS (DRYWELL + PRIMARY CONTAINMENT) AND THE EXTENT OF CORE DAMAGE IN REFERENCE PLANT



TITLE: REACTOR CORE DAMAGE ESTIMATION

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	
Small Liquid Line Break, Reactor Power ≥1%	Yes	---	Yes[1]	---	Yes[2]	
Small Liquid Line Break, Reactor Power <1%	---	---	Yes[1]	Yes	Yes[2]	RHR must be in shutdown cooling mode. 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]	RHR must be in shutdown cooling mode. 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]	Suppression pool must be in suppression cooling mode.
Large Liquid Line Break, Reactor Power <1%	---	Yes[4]	Yes[1]	Yes[3]	Yes[2]	RHR must be in shutdown cooling mode. Suppression pool must be in suppression cooling mode. Reactor water level must be raised and flow from moisture sepatators.

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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 ≥1%	Yes[3]	Yes[4]	---	---	Yes	
Large Steam Line Break, Reactor Power <1%	---	---	Yes[1]	Yes	Yes[2]	RHR must be in shutdown cooling mode. Reactor water level must be raised and flow from moisture separators.

1. Use if SRV's are vented to 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.

TITLE: REACTOR CORE DAMAGE ESTIMATION

PLANT PARAMETERS

PLANT	REACTOR TYPE/CONTAINMENT DESIGN	RATED POWER (MW_t)	PRIMARY COOLANT *		CONTAINMENT GAS**	
			REACTOR WATER MASS (10^8 g)	SUPPRESSION POOL WATER (10^9 g)	DRYWELL GAS VOL. (10^9 cc)	PRIMARY CONTAINMENT GAS VOLUME (10^9 cc)
CPS	BWR6/MKIII	2894	2.13	4.09	6.98	37.00

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

** Total Containment Gas Volume = Drywell Gas + Primary Containment

TITLE: REACTOR CORE DAMAGE ESTIMATION

SAMPLE CALCULATION OF FISSION PRODUCT INVENTORY CORRECTION FACTOR

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

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

where

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

λ_i = decay constant of nuclide i (day^{-1})

T_j = duration of operating period j (day)

T_j^0 = time between the end of operating period j and time of last reactor shutdown (day)

3651 = ave. operation power (in MWt) for the reference Station.

1095 = continuous operation time (in day) for the reference Station.

Assuming a reactor has the following power operation history:

Operation Period	Days since startup	Operation Time T_j (day)	T_j°	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

TITLE: REACTOR CORE DAMAGE ESTIMATION

For I-131 (λ = 0.0862 day $^{-1}$)

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

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

For Cs-137 (λ = 6.29 x 10^{-5} day^{-1})

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

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

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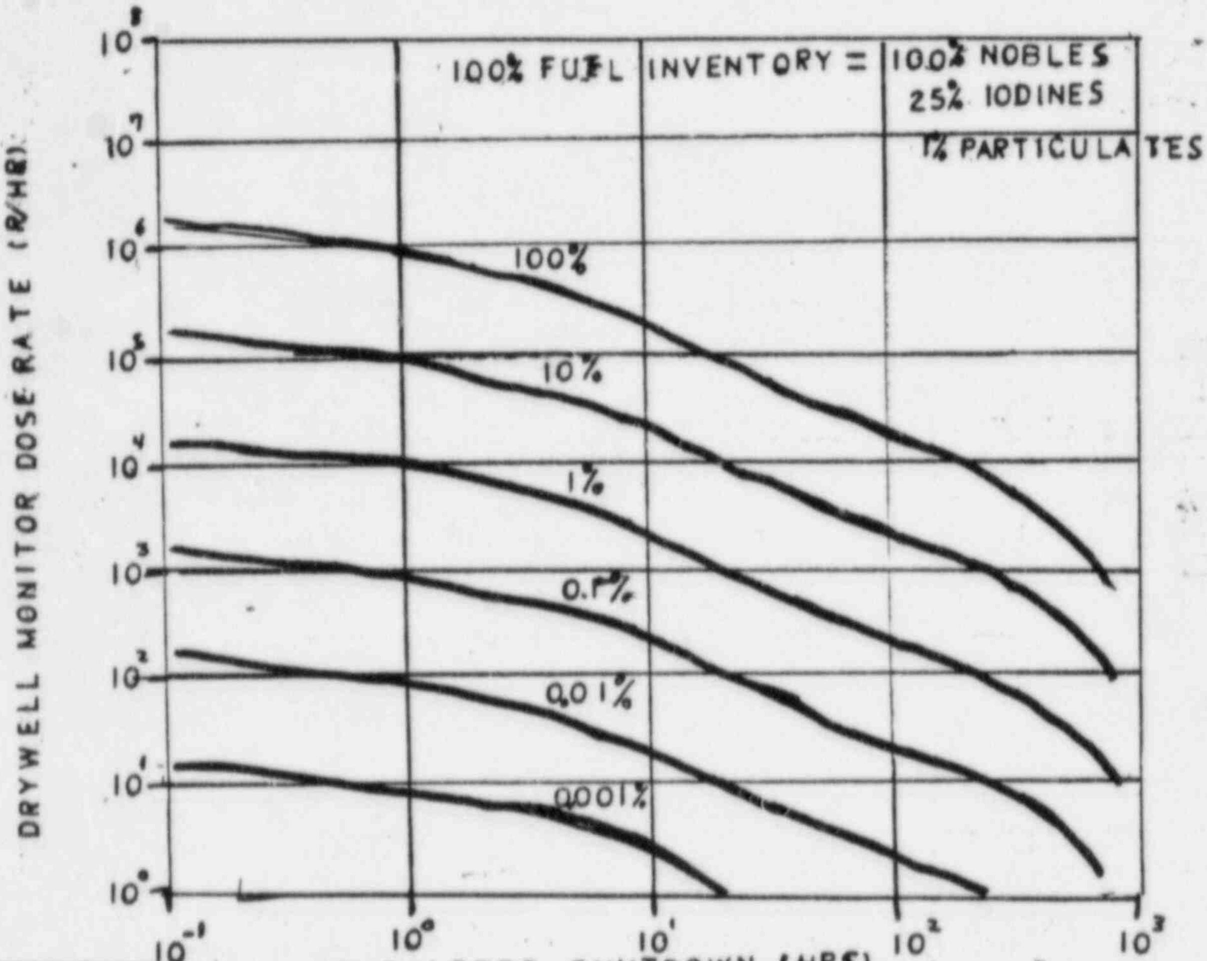
PERCENT OF FUEL INVENTORY AIRBORNE IN THE DRYWELL



% 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.

TITLE: REACTOR CORE DAMAGE ESTIMATION

MAXIMUM ACCEPTABLE CORE UNCOVERY TIME VS. TIME AFTER REACTOR SHUTDOWN