

RIVER BEND STATION
APPROVAL SHEET
STATION OPERATING PROCEDURES

NO. COP-1050

TITLE POST ACCIDENT ESTIMATION OF FUEL

CORE DAMAGE

SAFETY RELATED

YES

☒

NO

☐

TECHNICAL REVIEW REQUIRED

YES

☒

NO

☐

REV. NO.	PAGES ISSUED	INDEP. REVIEW SIGNATURE/DATE	TECH. REVIEW SIGNATURE/DATE	APPROVED BY SIGNATURE/DATE	EFFECT DATE
0	1 THRU 23 LATER'S PAGE	<i>E. Wash 27 Sep 84</i>	<i>P. M. M. 17/4/84</i>	<i>BE 10/18/84</i>	10/18/84
1	1 Thru 17 LATER'S PAGE	<i>E. Wash 15 Jan 85</i>	<i>PA 1/15/85</i>	<i>BE 1/15/85</i>	01/15/85
2	1 Thru 22	<i>E. Wash 13 May 85</i>	<i>PA 5-13-85</i>	<i>PA 5-13-85</i>	

FOR INFORMATION ONLY

CHEM

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RIVER BEND STATION
PROCEDURE (CHANGE) REVIEW FORM

PROC. NO. COP-1050

TITLE POST ACCIDENT ESTIMATION OF FUEL CORE DAMAGE

REV. NO. 2

REASON FOR PROCEDURE (CHANGE): Incorporation of Comments

YES

NO

☒

☒

IS THIS A NEW PROCEDURE

☐

☒

DOES THIS REVISION CHANGE THE SCOPE OF THE PROCEDURE

☐

☒

DOES THIS REVISION CHANGE THE INTENT OF THE PROCEDURE

☒

☒

IS THIS A REVISION TO A SAFETY RELATED PROCEDURE

PREPARED BY E.Dreyer

103-27-85

DATE

RESPON. SUPV. F. Lee Allis

PROCEDURE (CHANGE) CLASSIFICATION

YES

NO

☒

☐

SAFETY RELATED/QA APPLICABLE

☒

☐

TECHNICAL REVIEW REQUIRED

CLASSIFIED BY R. R. R. R. R.

TECH STAFF SIGNATURE

DATE 3/27/85

CROSS DISCIPLINE REVIEWS REQUIRED

☒

ALARA

☐

Operations

☐

Maintenance

☒

Tech Staff

☒

Chemistry

☐

ISI COORDINATOR

☒

QA

☐

RAD PRO

☐

Security

☐

APM-S

☐

APM-MAINT

☐

Radwaste

☒

APM-O/C/R

☐

OTHER

TECHNICAL REVIEW PAVob

DATE 5/13/85

RIVER BEND STATION
NUCLEAR SAFETY EVALUATION APPLICABILITY CHECK LIST/SAFETY EVALUATION
10CFR 50.59

- (1) PART A CHECK LIST APPLICABLE TO: COP 1050 Revision 2 TCN _____
(2) SAFETY EVALUATION APPLICABILITY CHECKLIST - PART-A

This procedure, procedure change or modification to which this evaluation is applicable represents:

- (2.1) YES _____ NO ☒ A change to the plant as described in the FSAR?
(2.2) YES _____ NO ☒ A change to procedures as described in the FSAR?
(2.3) YES _____ NO ☒ A test or experiment not described in the FSAR?
(2.4) YES _____ NO ☒ A change to the Technical Specification or Operating License?

If the answer to question 2.1, 2.2, 2.3 or 2.4 is "YES", complete ITEM (3). If the answer to all of the above is "NO" omit ITEM (3) (8) and (9)

(3) SAFETY EVALUATION - PART B

- (3.1) YES _____ NO _____ Will the probability of an accident previously evaluated in the FSAR be increased?
(3.2) YES _____ NO _____ Will the consequences of an accident previously evaluated in the FSAR be increased?
(3.3) YES _____ NO _____ May the possibility of an accident which is different than any already evaluated in the FSAR be created?
(3.4) YES _____ NO _____ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(3.5) YES _____ NO _____ Will the consequences of a malfunction of equipment important to safety different than any already evaluated in the FSAR be increased?
(3.6) YES _____ NO _____ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
(3.7) YES _____ NO _____ Will the margin of safety as defined in the basis to any Technical Specification be reduced?

If the answer to any of the above questions is "YES", an unreviewed safety question may be involved. If so, NRC review/approval prior to implementation is required (confer with licensing). Explain the basis for each answer (yes or no) on the SAFETY EVALUATION CONTINUATION SHEET.

- (4) REMARKS: (Attach additional pages if necessary) _____
(5) PREPARED BY: [Signature] NAME _____ DATE 13 May 85
(6) SUPV. REVIEWED BY: [Signature] NAME _____ DATE 5-13-85
(7) TECH STAFF REVIEW (FIG. 1): [Signature] NAME _____ DATE 5/13/85
(8) FRC APPROVAL: _____ NAME _____ DATE _____
(9) NRB REVIEW: _____ NAME _____ DATE _____

POST ACCIDENT ESTIMATION OF
FUEL CORE DAMAGE

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FUEL CORE DAMAGE

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1.0 PURPOSE/SCOPE/APPLICABILITY

- 1.1 The purpose of this procedure is to provide a preliminary estimate of the reactor core damage from the measured fission product concentration, under accident conditions.
- 1.2 The procedure involves calculation of fission product inventories of the primary system under postulated loss of coolant accident conditions.
- 1.3 A BWR-6 with a Mark III containment is used as a reference plant. Application of reference plant core damage graphs to the RBS reactor is possible by applying pertinent correction factors discussed in the procedure.
- 1.4 The fuel gap fission products are assumed to be released upon the rupture of fuel cladding.
- 1.5 The majority of fission product inventories in the fuel rods would be released when the fuel is melted at high temperatures.
- 1.6 The fission product concentrations are obtained by sampling the primary coolant and/or the containment atmosphere via post accident sampling system (PASS).
- 1.7 The Chemistry/Core Damage Assessment Coordinator will perform the core damage assessment estimate and report the results to the Technical Support Center Emergency Director.

2.0 REFERENCES

- 2.1 NEDO-22215 82NED090, Procedure for the Determination of the Extent of Core Damage, Under Accident Conditions
- 2.2 USNRC Reg Guide 1.97, Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, 1980
- 2.3 NEDO-30088, Responses to NRC Post-Implementation Review Criteria for Post-Accident Sampling System
- 2.4 RBS FSAR, Section 13.3.5.2, Emergency Planning Assessment Actions
- 2.5 RBS FSAR, Volume 1, Chapter 1.1, Introduction and General Description of Plant
- 2.6 RBS FSAR, Volume 8, Table 4.2-4, Fuel Data and Table 4.3-1 Reactor Core Dimensions

- 2.7 US Atomic Energy Commission, Safety Evaluation of the River Bend Station, Sept. 1974
- 2.8 EIP-2-015, Post-Accident Sampling Operations
- 2.9 CORDAM.BAS, Chemistry Computer Routine
- 2.10 COP-0425, Determination of the H₂ and O₂ Gas - Gas Chromatography Method
- 2.11 COP-1001, Post-Accident Sampling of Primary Coolant
- 2.12 COP-1002, Post-Accident Sampling of Containment Atmosphere
- 2.13 COP-1030, Post-Accident Isotopic Analysis for Particulate, Iodine and Gaseous Activity
- 2.14 COP-1033, Post-Accident Isotopic Analysis for Liquid Activity

3.0 DEFINITIONS

3.1 Metal-Water Reaction - The reaction between Zircaloy from the fuel cladding and the water under accident conditions is called Metal-Water reaction. % Metal-Water reaction represents % cladding failure.

3.2 PAS - Post-Accident Sample

4.0 REQUIRED EQUIPMENT/CHEMICALS

N/A

5.0 PRECAUTIONS

N/A

6.0 LIMITATIONS

- 6.1 The performed estimates are done under the presumption that no reactor coolant cleanup systems are operated after the accident.
- 6.2 Measurement of Cs-137 and Kr-85 activities may not be possible until the reactor has been shut down for several weeks to allow the decay of shorter lived isotopes.
- 6.3 If isotopes of Sr, Ba, La and Ru are found in significant concentrations some degree of fuel melting may be inferred. However, the extent of fuel melting cannot be determined based on the concentrations of these nuclides because of the lack of baseline data.

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- 6.4 Containment hydrogen concentration measurements via PASS or containment gas analyzers provides a measure of the extent of metal-water reaction which is used as an estimate of cladding damage.
- 6.5 Core damage below 1 % is assumed to be a non-accident condition in view of cladding failure.

7.0 PREREQUISITES/INITIAL CONDITIONS

- 7.1 Post-Accident sampling and analysis completed per References 2.10, thru 2.14.

8.0 PROCEDURE/INSTRUCTIONS/REQUIREMENTS

8.1 Sequence of Analysis

- 8.1.1 Attachment 5 shows as an example the sequence of steps to assess the extent of damage to the core, after its possibility has been established. Indicators for core damage or its possibility are increased radiation levels (i.e. Off-Gas Pre-Treatment, Main Steam Line, Drywell Post Accident Area Monitor) or decreased water levels (i.e., Reactor water level less than 160")
- 8.1.2 The sequence of analysis from Attachment 5 may be changed by the Chemistry/Core Damage Assessment Coordinator according to the availability of data and/or information about the extent and course of the accident which caused the core damage. (Subsections of this procedure may therefore be performed out of sequence.)

8.2 Selection of Sample Point

NOTE

Depending on the actual extent of the accident, the Chemistry/Core Damage Assessment Coordinator might require the analysis of additional samples taken from the Drywell and/or Suppression Pool, under consideration of the respective volumes (Item 10.1, Section 10) to correct the individual fission product concentration results of the reactor coolant or the containment atmosphere.

- 8.2.1 Refer to the following table for selecting the recommended sample location for gas samples depending on the event type causing the core damage:

<u>Event Type</u>	<u>Sample Location</u>
Non-Breaks (e.g., MSIV Closure)	Containment Atmosphere
Small Breaks	Drywell (before depress.) Containment Atmosphere (after depress.)
Large Breaks (liquid or steam) in Containment	Drywell
Large Breaks outside containment	Containment Atmosphere

- 8.2.2 For liquid sampling per Reference 2.2, the optimum sample point for all events is the jet pumps as long as there is sufficient reactor pressure to provide a sample location. If there is not sufficient reactor pressure to allow a sample to be taken from the jet pumps, then the sample should be taken from the sample point in the RHR system.

NOTE

Refer to Attachment 3 for explanation of symbols.

- 8.2.3 If separate measurements of the activity concentration are made in the containment and drywell atmosphere, the resultant average concentration is calculated for RBS by using:

$$C_{gi} = C_{Drywell} \times 0.174 + C_{Cont.} \times 0.826$$

- 8.2.4 Fission product concentration correction for temperature and pressure difference in sample vial (T_v , P_v) and in containment (T_c , P_c) are :

$$C_{gi}(Cont) = C_{gi}(vial) \times \frac{P_c T_v}{P_v T_c}$$

- 8.2.5 If separate measurements of the specific activity are made for the suppression pool and reactor water, the resultant average concentration is calculated for RBS by using:

$$C_{wi} = C_{Rx} \times 0.053 + C_{supp} \times 0.847$$

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8.3 Determination of Core Damage from Fission Products Concentration

- 8.3.1 Obtain primary coolant and/or containment atmosphere sample from the post-accident sampling system per Reference 2.11 and/or Reference 2.12 and record sample time on Attachment 1.
- 8.3.2 Obtain fission products concentration in the samples per Reference 2.13. Record concentrations of I-131, Cs-137, Xe-133 and Kr-85 on Attachment 1.

NOTE

Calculations in Steps 8.3.3 to 8.3.6 are performed per computer by using Reference 2.9.

- 8.3.3 Record the shutdown time on Attachment 1 and correct measured concentrations for decay to the time of reactor shutdown.
- 8.3.4 Calculate the fission products inventory correction factors as per Attachment 2.
- 8.3.5 By using the correction factors determined in Sections 10.2 thru 10.4, calculate the normalized concentrations of the fission products as per Attachment 3.
- 8.3.6 Use Figures 1 through 4 (Attachments 6 through 9) to estimate the extent of fuel or cladding damage. Record the results of the estimated damage on the Attachment 4.

8.4 Determination for Cladding Damage from Hydrogen Concentration in the Containment Atmosphere

- 8.4.1 Determine hydrogen concentration in the containment atmosphere post accident sample as % H_2 (dry basis) per Reference 2.10. The Containment Hydrogen monitor reading may be used as an alternative.
- 8.4.2 Using a curve for Mark III Containment in Fig. 5 (Attachment 10), determine the cladding damage (% metal-water reaction) for the reference plant.

- 8.4.3 The % metal water reaction (% MW) for RBS is assumed to be the same as the % metal water reaction for the reference plant (% MW Ref.) within the accuracy of estimate, since:

$$\% \text{ MW} = \% \text{ MW}_{\text{ref}} \times \frac{748}{N} \times \frac{V}{1.36 \text{ E6}}$$

$$= \% \text{ MW}_{\text{ref}} \times 1.05$$

where = N = RBS number of fuel bundles = 624

V = RBS Containment Air Volume = 1.443 E6 ft³

- 8.4.4 Record the results of estimated damage from Step 8.4.2 on Attachment 4.

8.5 Determination of Core Damage from the Containment Radiation Level

NOTE

This method can only be used if substantial core damage and release of fission products into the containment has occurred.

- 8.5.1 Obtain reading of drywell radiation monitor, (R) in R/hr.
- 8.5.2 Determine elapsed time from plant shutdown to the containment radiation monitor reading (t) in hours.
- 8.5.3 Using Attachment 11, determine the fuel inventory release for the reference plant (I)_{ref} in %.
- 8.5.4 Determine the inventory release to the containment (I) in % using the following formula:

$$(I) = (I)_{\text{ref}} \cdot \frac{1670}{P} \left(\frac{V}{237,450} \right) (6/D)$$

$$= (I)_{\text{ref}} \times \frac{6.09 \text{ E4}}{P}$$

where P = reactor power level, MW_{th}

V = total containment free volume, 1.443 E6 ft³

D = distance of detector from reactor biological shield wall, 19.6 ft

- 8.5.5 Record the result of Step 8.5.4 on Attachment 4.

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8.5.6 Retain all pertinent data, calculations, etc. for permanent retention.

9.0 RESTORATION

N/A

10.0 DATA REQUIREMENTS

10.1 River Bend Station General Information

10.1.1 MWT - 2894

10.1.2 Assemblies - 624

10.1.3 Fuel Rods/Assembly - 62

10.1.4 Water Rods/Assembly - 2

10.1.5 Reactor Coolant Mass 1.99 E8 gm

10.1.6 Supression Pool Volume - 126,600 ft³ (3.57 E9 ml)

10.1.7 Total Volume Reactor Cool. + Supr. Pool - 3.77 E9 ml

10.1.8 Containment Net Free Ai Volume - 1.192E6 ft³ (3.38 E10 ml)

10.1.9 Drywell Net Free Air Volume - 2.51E5 ft³ (7.11 E9 ml)

10.1.10 Cont. + Drywell Net Free Air Vol. - 1.443E6 ft³ (4.09 E10ml)

10.2 Inventory Correction Factor (F_{Ii})

$$F_{Ii} = \frac{\text{Inventory of Nuclide } i \text{ in the Reference Reactor Core}}{\text{Inventory of Nuclide } i \text{ in the RBS Reactor Core}}$$

10.3 Containment Gas Volume Correction Factor (F_g)

$$F_g = \frac{\text{RBS containment gas volume, (4.09 E10 cc)}}{\text{Reference plant Containment Gas Volume (4.0 E10 cc)}} = 1.02$$

10.4 Primary Coolant Mass Correction Factor (F_w)

$$F_w = \frac{\text{Operation Plant Coolant Mass (3.77 E9 gm)}}{\text{Reference Plant Coolant Mass (3.92 E9 gm)}} = 0.96$$

11.0 ACCEPTANCE CRITERIA

N/A

"END"

Fission Product ConcentrationsShutdown Time, T_0 : _____ hrs , Date _____Sample Time, T_s : _____ hrs , Date _____Decay Time, $T = (T_s - T_0)$: _____ hrs \div 24 = _____ day* C_i = Fission Product Concentration of the Post Accident Sample. λ_i = Decay Constant of the nuclide, day^{-1} * C_{iT} = Decay Corrected Fission Product Concentration.

$$C_{iT} = C_i e^{\lambda_i T}$$

FISSION PRODUCT	C_i	λ_i	T	$e^{\lambda_i T}$	C_{iT}
I - 131		8.62 E-2			
Cs - 137		6.29 E-5			
Xe - 133		1.32 E-1			
Kr - 85		1.77 E-4			

* concentrations expressed in uCi/gm for liquids and uCi/ml for gases.

Chemistry Core Damage Assessment Coordinator _____

Date _____

Time _____

Inventory Correction Factors (F_{Ii})Equation 1

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

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

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 and time of the final reactor shutdown, days λ_i = Decay correction factor of the isotope, days⁻¹

3651 = Avg. Operation Power (in MWt) for the reference plant

1095 = Continuous Operation time (in Days) for the reference plant

Using Equation 1, compute F_{Ii} for the fission products using λ_i values provided in the following table. A sample calculation is shown on Page 2 of Attachment 2. Attach Calculation Sheets.

FISSION PRODUCT	λ_i	INVENTORY CORRECTION FACTOR, F_{Ii}
I-131	8.62 E-2	
Cs-137	6.29 E-5	
Xe-133	1.32 E-1	
Kr-85	1.77 E-4	

Sample Calculation of Fission Product Inventory Correction Factors, F_{Ii}

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 (= 314 - 60)	1000
1B	61 - 70	---	---	0
2A	71 - 270	200	44 (= 314 - 270)	2000
2B	271 - 300	---	---	0
3	301 - 314	14	0 (= 314 - 314)	3000

* For I-131 ($\lambda = 0.0862 \text{ day}^{-1}$)

$$F_{I(I-131)} = \frac{3651(1-e^{-0.0862 \times 1095})}{1000(1-e^{-0.0862 \times 60})e^{-0.0862 \times 254} + 2000(1-e^{-0.0862 \times 200})e^{-0.0862 \times 44} + 3000(1-e^{-0.0862 \times 14})e^{-0.0862 \times 0}}$$

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

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

$$F_{I(Cs-137)} = \frac{3651(1-e^{-6.29 \times 10^{-5} \times 1095})}{1000(1-e^{-6.29 \times 10^{-5} \times 60})e^{-6.29 \times 10^{-5} \times 254} + 2000(1-e^{-6.29 \times 10^{-5} \times 200})e^{-6.29 \times 10^{-5} \times 44} + 3000(1-e^{-6.29 \times 10^{-5} \times 14})e^{-6.29 \times 10^{-5} \times 0}}$$

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

Chemistry Core Damage Assessment Coordinator

Date

Time

1. Normalized Concentration of Primary Coolant Fission Products, C_{wi}^{Ref}

$$= C_{it} \times F_{Ii} \times F_w \quad (A)$$

$$= C_{it} \times F_{Ii} \times 0.96$$

1.1 Normalized Concentration of I-131

$$C_{it} = \text{_____} \text{ uCi/gm (From Attach. 1)}$$

$$F_{Ii} = \text{_____} \text{ (From Attach. 2)}$$

Substituting in equation (A) for I-131,

$$\begin{aligned} C_{wi}^{Ref} &= \text{_____} \times \text{_____} \times 0.96 \\ &= \text{_____} \text{ uCi/gm} \end{aligned}$$

1.2 Normalized Concentration of Cs-137

$$C_{it} = \text{_____} \text{ uCi/gm (From Attach. 1)}$$

$$F_{Ii} = \text{_____} \text{ (From Attach. 2)}$$

Substituting in equation (A) for Cs-137

$$\begin{aligned} C_{wi}^{Ref} &= \text{_____} \times \text{_____} \times 0.96 \\ &= \text{_____} \text{ uCi/gm} \end{aligned}$$

2. Normalized Concentration of Containment Atmosphere Fission Products, C_{gi}^{Ref}

$$= C_{it} \times F_{Ii} \times F_g \quad (B)$$

$$= C_{it} \times F_{Ii} \times 1.02$$

2.1 Normalized Concentration of Xe-133

$$C_{it} = \text{_____} \text{ uCi/ml (From Attach. 1)}$$

$$F_{Ii} = \text{_____} \text{ (From Attach. 2)}$$

Substituting for Xe-133, in equation (B).

$$\begin{aligned} C_{gi}^{Ref} &= \text{_____} \times \text{_____} \times 1.02 \\ &= \text{_____} \text{ uCi/ml} \end{aligned}$$

2.2 Normalized Concentration of Kr-85

$$C_{it} = \text{_____} \text{ uCi/ml (From Attach. 1)}$$

$$F_{Ii} = \text{_____} \text{ (From Attach. 2)}$$

Substituting for Kr-85, in equation (B).

$$\begin{aligned} C_{gi}^{\text{Ref}} &= \text{_____} \times \text{_____} \times 1.02 \\ &= \text{_____} \text{ uCi/ml} \end{aligned}$$

Record the calculated normalized concentrations in the box below and use the information to compute core damage using Figures 1 through 4, (Attachments 5 through 8).

I-131	_____	uCi/gm
Cs-137	_____	uCi/gm
Xe-133	_____	uCi/ml
Kr-85	_____	uCi/ml

Chemistry Core Damage Assessment Coordinator

Date

Time

1. Core Damage Estimates based in the Fission Products Concentrations in the post accident samples

Record results obtained based on concentrations reported in Attachment 3 and using Figures 1 through 4 in the following table.

FISSION PRODUCT	% CLADDING FAILURE	% FUEL MELTDOWN
I-131		
Cs-137		
Xe-13		
Kr-'		

2. Cladding failure estimate based on the Hydrogen Concentration in the containment

Record results obtained per Step 8.4.2:

_____ % MW reaction
(% Cladding Failure)

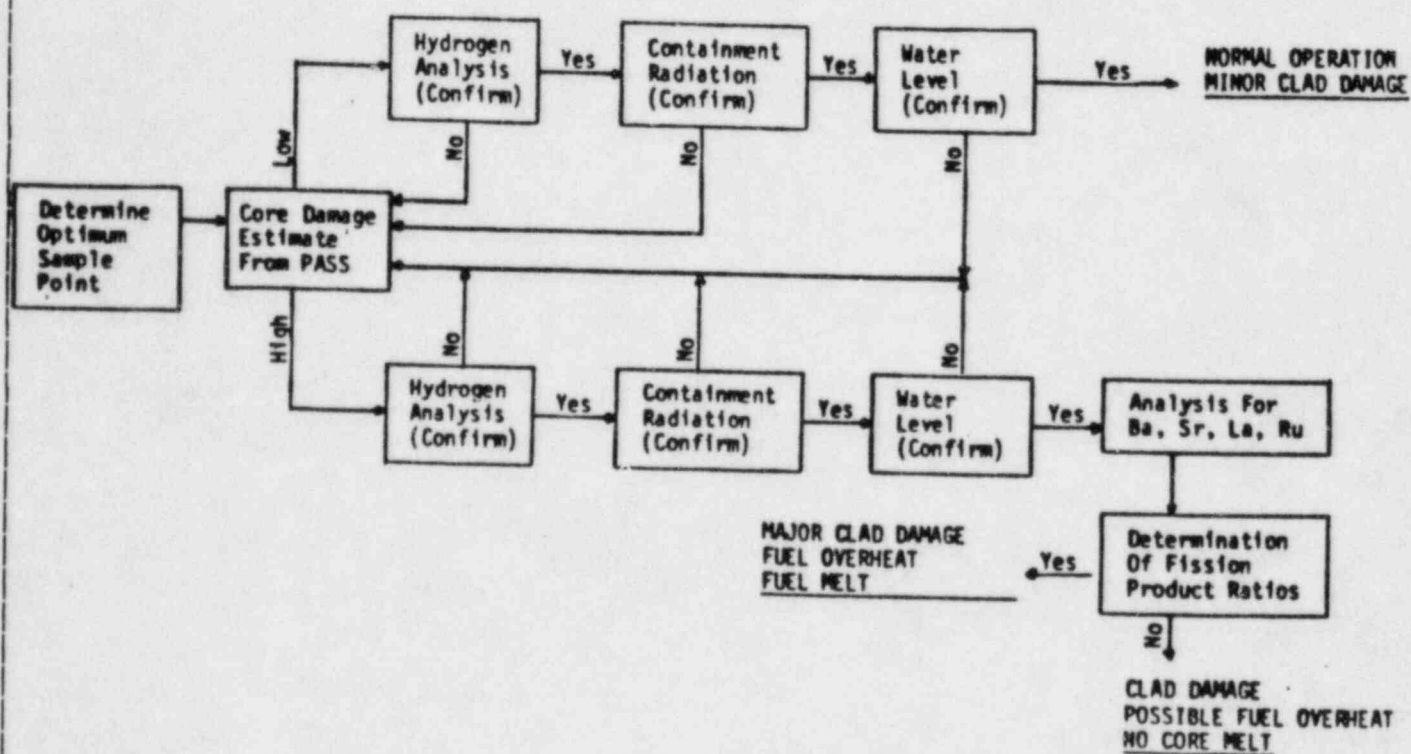
3. Core damage estimate based on radiation level in the drywell caused by airborne fuel inventory of fission products per Step 8.5.5

_____ % Core Damage
(% Inventory airborne)

Chemistry Core Damage Assessment Coordinator

Date

Time



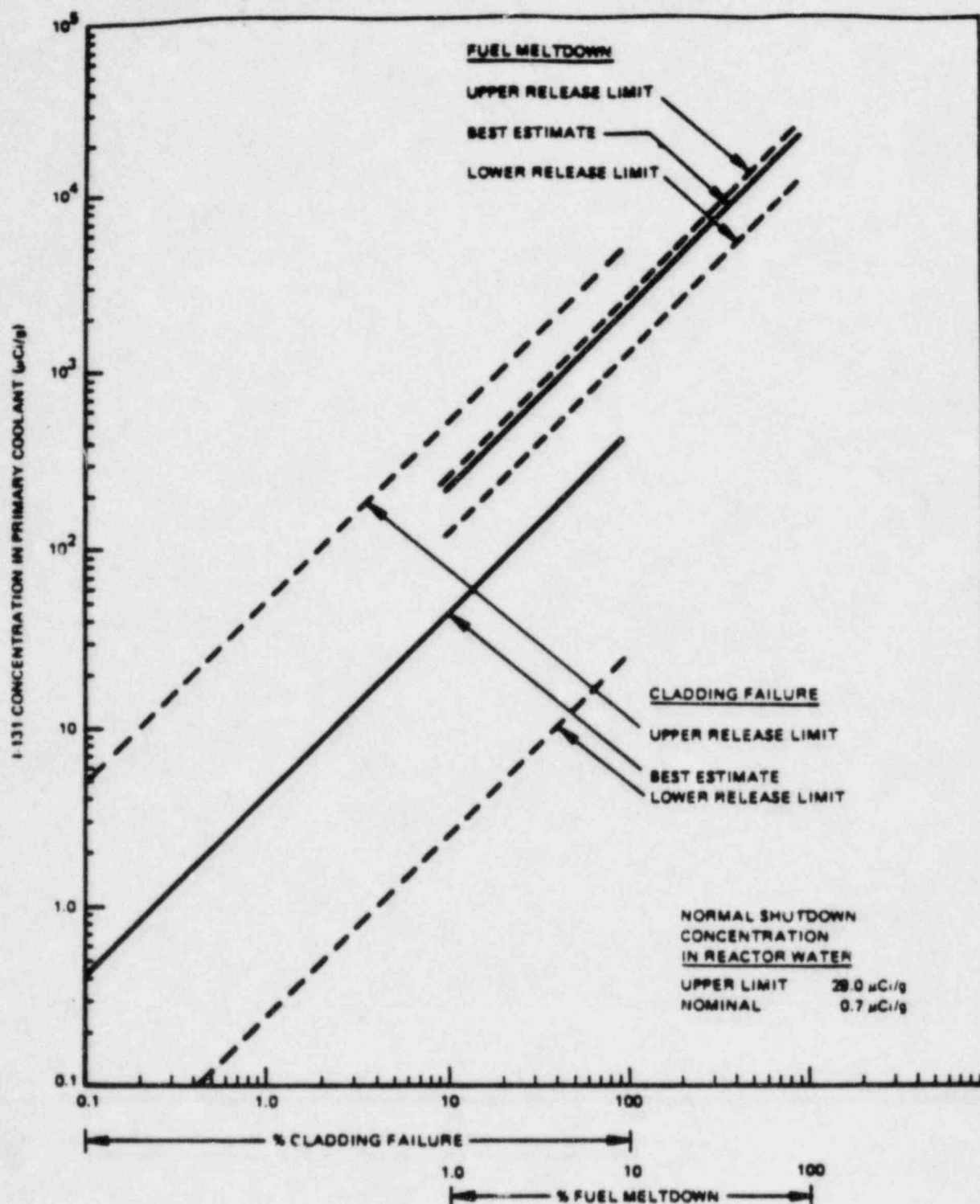


Fig. 1 I-131 Concentration in the Primary Coolant vs the Extent of Core Damage in the Reference Plant

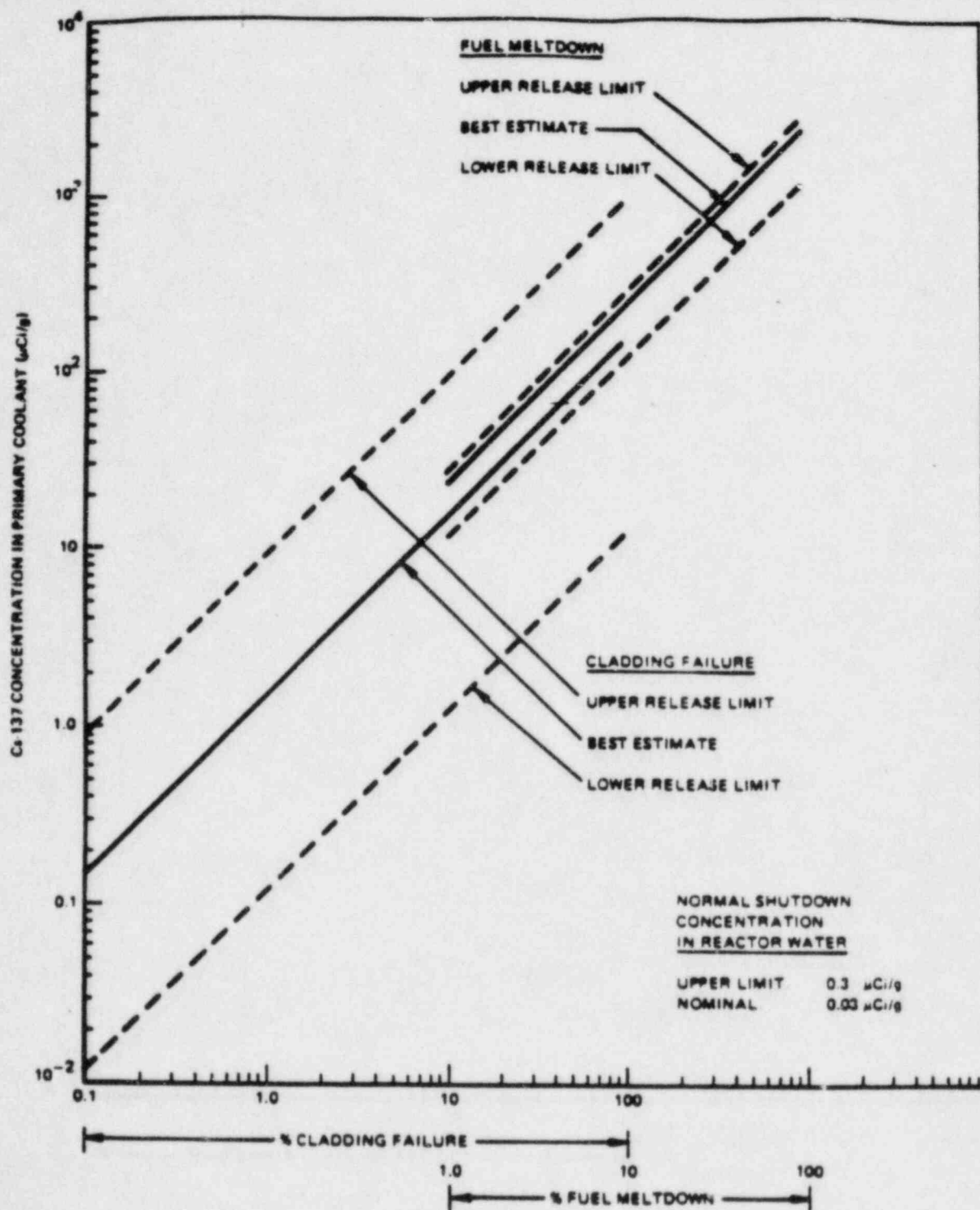


Fig. 2 Cs-137 Concentration in the Primary Coolant vs the Extent of Core Damage in the Reference Plant

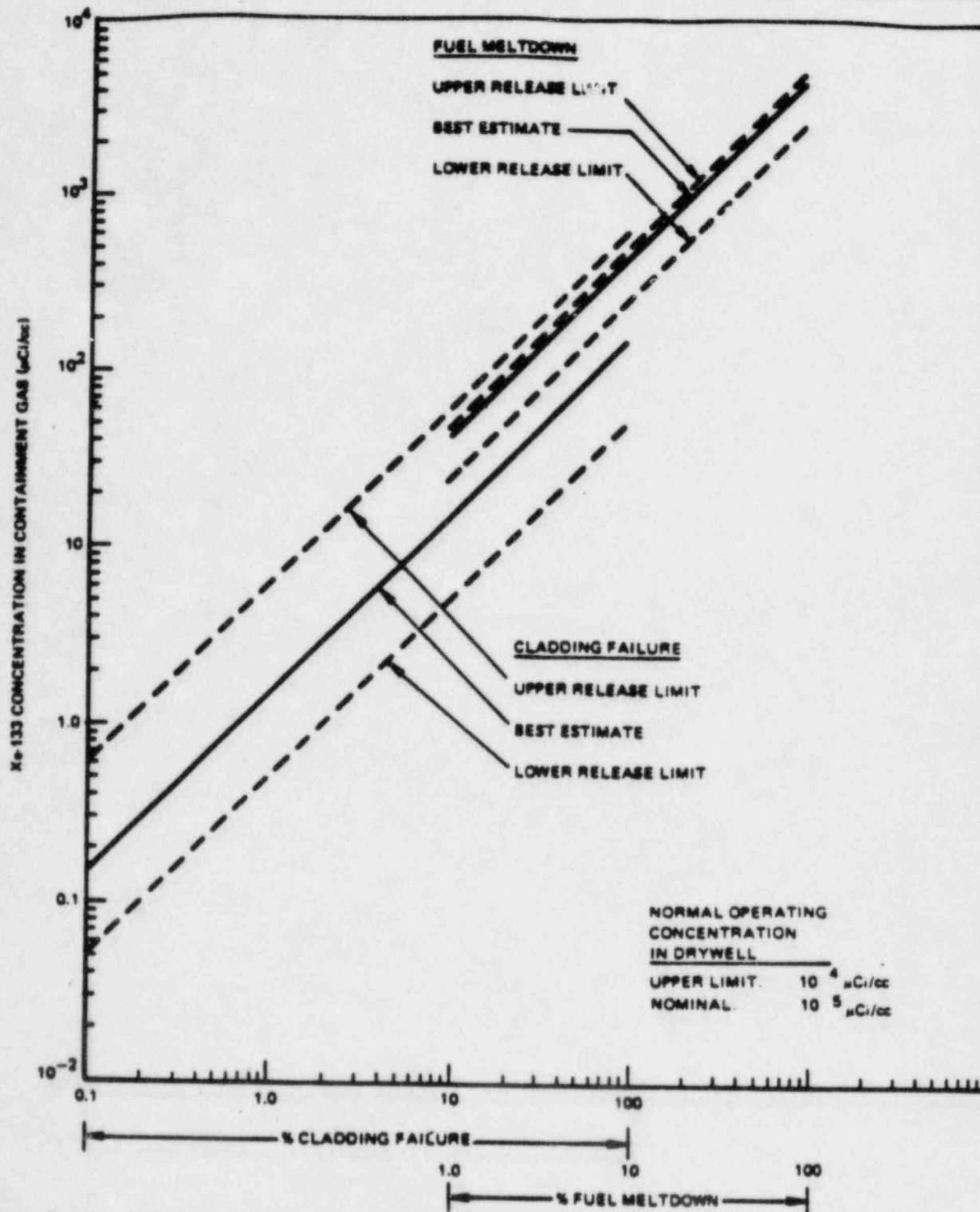


Fig. 3 Xe-133 Concentration in the Containment vs the Extent of Core Damage in the Reference Plant

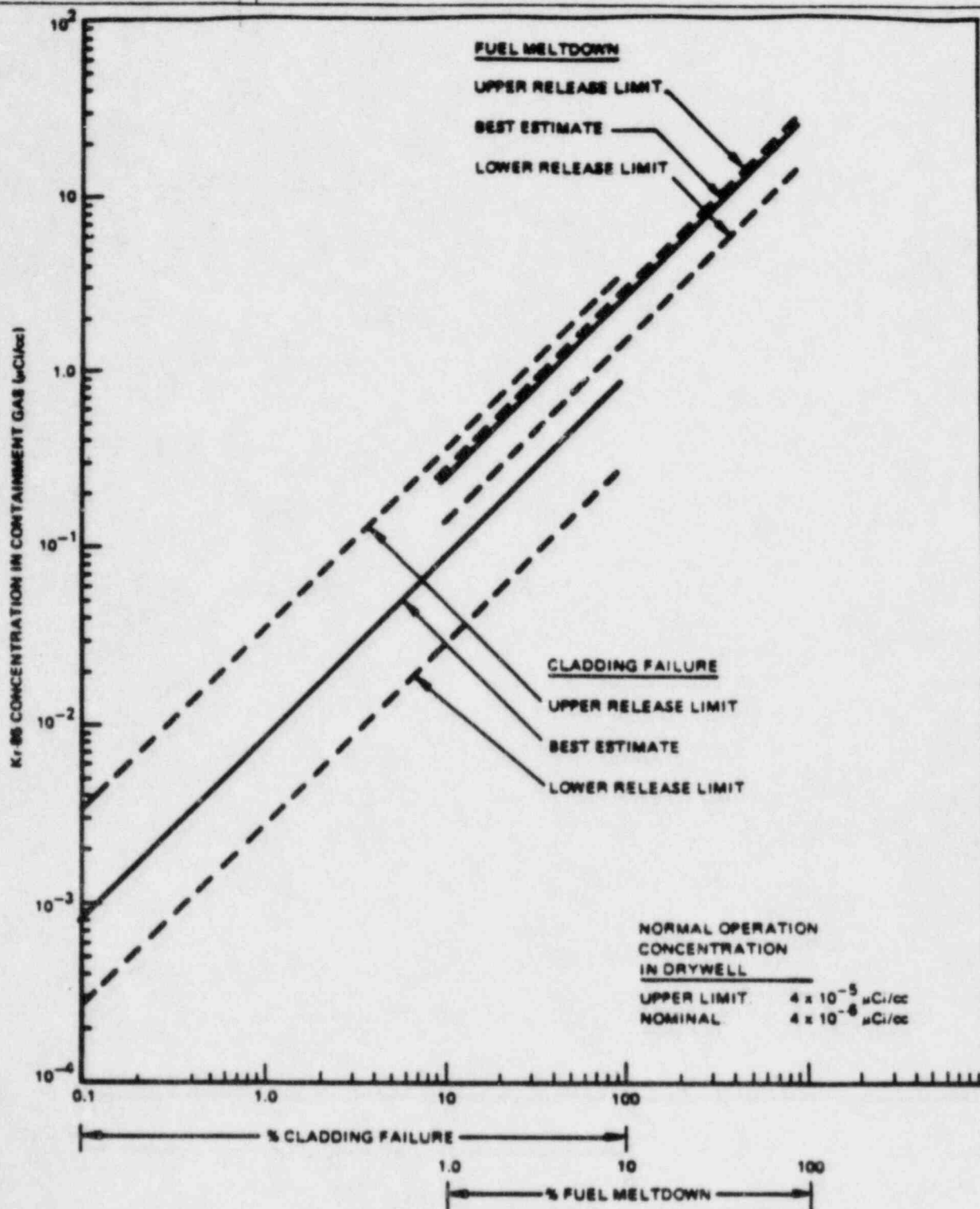


Fig. 4 Kr-85 Concentration in the Containment vs the Extent of Core Damage in the Reference Plant

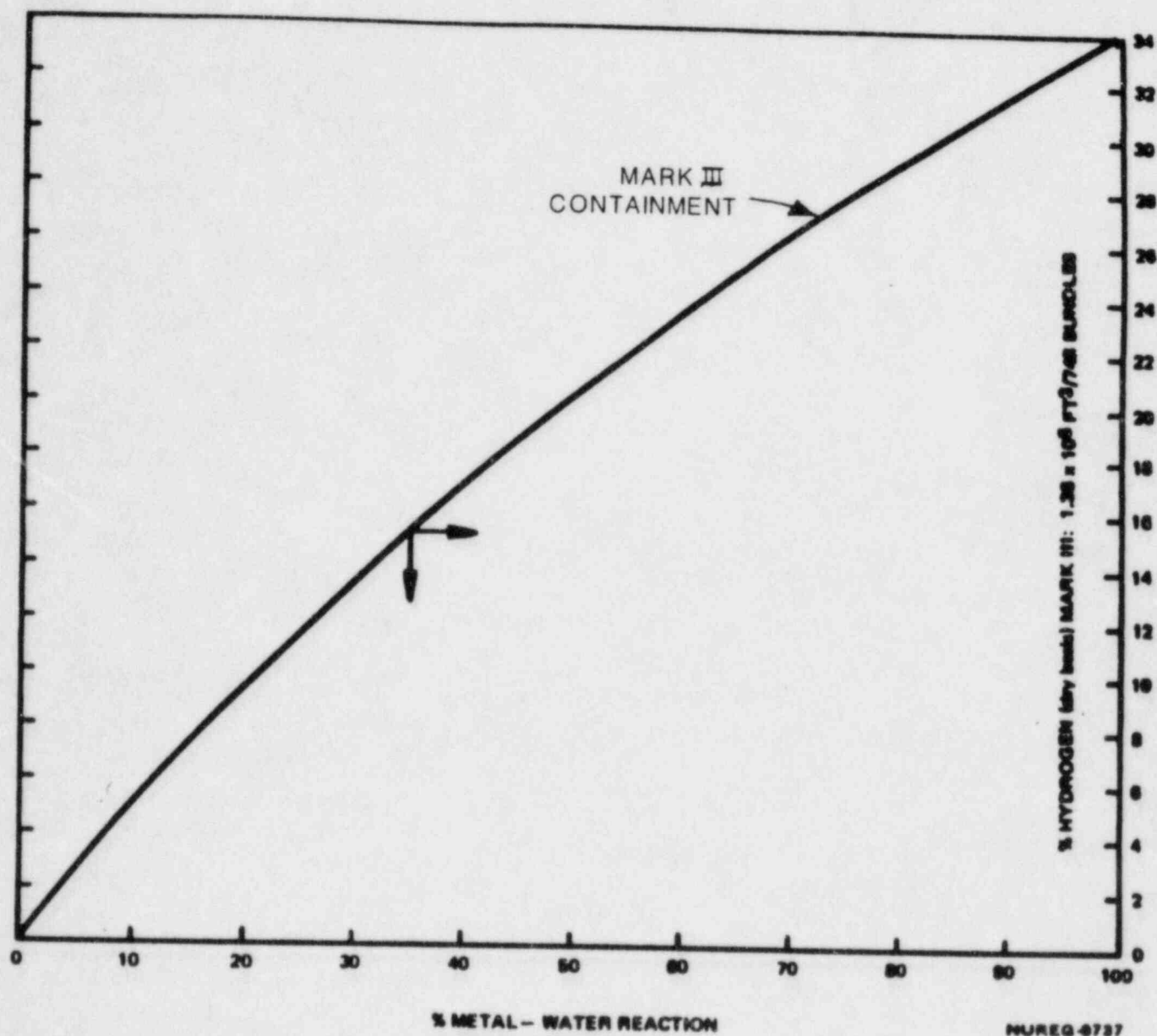
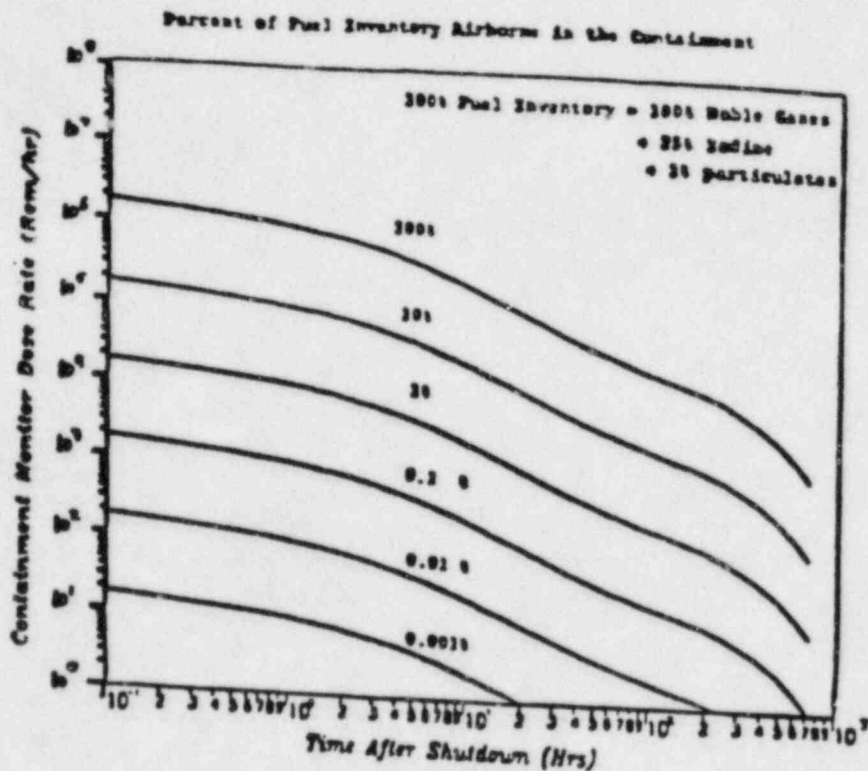


Fig. 5 Hydrogen Concentration in the Containment vs % Metal Water Reaction
(% Cladding Failure)

CONTAINMENT AIRBORNE ACTIVITY FROM
RADIATION LEVEL

% Fuel
Inventory
Released

Approximate Source and Damage Estimate

100.	100% TID-14844, 100% fuel damage, potential core melt.
10.	10% TID noble gases, HMI source.
1.	1% TID, 10% HMI gap activity, total clad failure, partial core uncovered.
0.1	0.1% TID, 10% HMI gap activity, major clad failure.
0.01	0.01% TID, 10% HMI gap, Max. 10% clad failure.
0.001	0.001% TID, 10% HMI gap, 10% clad failure, local heating of 2-10 fuel assemblies.
10^-2	0.01% TID, 0.01% HMI gap, clad failure of 2/4 fuel elements (26 rods).
10^-3	0.01% HMI gap, clad failure of a few rods.
10^-4	100% coolant release with spiking.
10^-5	100% coolant inventory release.
10^-6	Upper range of normal airborne noble gas activity in containment.

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