ENCLOSURE 4

RBS Procedure COP-1050

"Post-Accident Estimation of

Fuel Core Damage"

RIVER BEND STATION
APPROVAL SHEET
STATION OPERATING PROCEDURES

NO. COP-1050 **TITLE** POST ACCIDENT ESTIMATION OF FUEL CORE DAMAGE

SAFETY RELATED YES [X] NO [ ]

TECHNICAL REVIEW REQUIRED YES [X] NO [ ]

REV. NO.	PAGES ISSUED	INDEP. REVIEW SIGNATURE/DATE	TECH. REVIEW SIGNATURE/DATE	APPROVED BY SIGNATURE/DATE	EFFECT DATE
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PROC. NO. COP-1050 TITLE POST ACCIDENT ESTIMATION OF FUEL CORE DAMAGE

REV. NO. 0

The following is a list of "missing" and/or "tentative information" -- (LATER'S) -- in this procedure. The responsible Section Supervisor shall update this procedure as the information becomes available.

(LATER) LOCATION SECTION/STEP	PAGE	BRIEF DESCRIPTION
3.2	2	COP-1001 NOT ISSUED
3.3	2	COP-1002 NOT ISSUED
3.9	2	COP-0425 NOT ISSUED
3.11	2	COP-1033 NOT ISSUED
3.13	2	COP-1030 NOT ISSUED
4.1.4	3	REACTOR COOLANT VOLUME
4.1.8	3	TOTAL MASS OF ZIRCONIUM
ATTACHMENT 3	9	TOTAL MASS OF ZIRCONIUM
ATTACHMENT 4	10	REACTOR COOLANT AND SUPPRESSION POOL VOLUME
ATTACHMENT 4	10	TOTAL ACTIVITY OF I-131
ATTACHMENT 5	11	VOLUME OF REACTOR COOLANT AND SUPPRESSION POOL
ATTACHMENT 5	11	TOTAL ACTIVITY OF CS-137
ATTACHMENT 6	12	TOTAL ACTIVITY OF XE-133
ATTACHMENT 7	13	TOTAL ACTIVITY OF KR-85

POST ACCIDENT ESTIMATION OF FUEL CORE DAMAGE

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1.0 PURPOSE

1.1 The purpose of this procedure is to provide the method and calculations necessary to estimate fuel core damage from a major reactor accident.

2.0 DISCUSSION

2.1 This procedure provides a quick first estimate of the extent of fuel core damage based on the methods outlined in Reference 3.8. A direct method, employing specific RBS data is covered in Subsection 8.1 to 8.6. An indirect method, using plant perimeter correction factors which relate RBS to the Reference Plant (GE Standard 3579 MW, BWR 6/III) is outlined in Subsection 8.7. This last method utilizes standardized graphs of nuclide concentration vs core damage supplied with Reference 3.8 (Attachments 9 to 12) to estimate the extent of core damage.

2.2 Calculation for the determincation of percent fuel damage shall be performed by the Sampling Team and reviewed by the Chemistry/Core Damage Assessment Coordinator.

2.3 Analytical results are obtained by sampling the Containment and/or Drywell Atmosphere, Supression Pool and/or Reactor Coolant System.

2.4 The Chemistry Core Damage Assessment Coordinator will inform the Technical Support Center Emergency Director with the percent fuel damage results as soon as the information is available and verified.

2.5 Measurements 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 the shorter lived isotopes.

3.0 REFERENCES

3.1 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

3.2 COP-1001, Post-Accident Sampling of Primary Coolant (LATER)

3.3 COP-1002, Post-Accident Sampling of Containment Atmosphere (LATER)

3.4 RBS FSAR, Section 13.3.5.2, Emergency Planning Assessment Actions

3.5 RBS FSAR, Volume 1, Chapter 1.1, Introduction and General Description of Plant

3.6 RBS FSAR, Volume 8, Table 4.2-4, Fuel Data

3.7 US Atomic Energy Commission, Safety Evaluation of the River Bend Station, Sept. 1974

3.8 NEDO-22215, 82NED090, Procedures for the Determination of the Extent of Core Damage Under Accident Conditions.

3.9 COP-0425, Determination of the H_2 and O_2 gas - Gas Chromatography Method.

3.10 NEDC-30088, Responses to NRC Post-Implementation Review Criteria for Post-Accident Sampling System

3.11 COP-1033, Post-Accident Isotopic analysis for Liquid Activity (LATER).

3.12 EIP-2-015, Post-Accident Sampling Operations

3.13 COP-1030, Post-Accident Isotopic Analysis for Particulate/ Iodine/Gaseous Activity

4.0 DEFINITIONS

4.1 River Bend Station General Information

4.1.1 MWT - 2894

4.1.2 Assemblies - 624

4.1.3 Fuel Rods/Assembly - 62

4.1.4 Reactor Coolant Volume (LATER)

4.1.5 Supression Pool Volume - 126,600 ft^3 (3.57 E9 cc)

4.1.6 Containment Net Free Air Volume - 1.120E6 ft^3 (3.17 E10 cc)

4.1.7 Drywell Net Free Air Volume - 2.47E5 ft^3 (6.99 E9 cc)

4.1.8 Containment + Drywell Net Free Air Volume - 1.367 ft^3 (3.87 E10cc)

4.1.9 Zirconium - (LATER) lb. total

4.2 Hydrogen burn - The initiation of the hydrogen igniters to decrease the hydrogen concentration in containment.

4.3 Evidence of Hydrogen Burn - Evidence of hydrogen burn is depicted by an increase of condensation and a drop of pressure in containment.

5.0 PRECAUTIONS/LIMITATIONS

5.1 Iodine and Xenon analysis are based upon equilibrium fuel power isotopic concentrations. If fuel damage is suspected to have occurred during times of reduced power or near the time of significant power change, the Iodine and Xenon inventory must be compensated accordingly. Refer to Attachment 1 to calculate the power correction factor Y.

5.2 Cs-137 and Kr-85 concentrations will be corrected by multiplying the average capacity factor for the previous 3 years due to their long half lives. Refer to Attachment 1 to calculate the power correction factor Z.

5.3 Core damage below 1% is assumed to be a non-accident condition.

5.4 If the isotopic analysis show the absence of Ruthenium and Tellurium, then assume that fuel melting has not occurred. However, the presence of these nuclides does not necessarily confirm fuel melting.

5.5 The determination of percent failed fuel is highly dependent on core temperature reached during the accident condition. Core temperature in excess of 1600^{o}F indicate possible cladding damage. Temperatures in excess of 4000^{o}F indicate possible fuel melting.

5.6 The reactor coolant temperature shall be compensated by a Density Correction Factor. Refer to Attachment 2.

5.7 The performed estimates are done under the presumption that no reactor coolant cleanup systems are operated after the accident.

6.0 REQUIRED EQUIPMENT/MATERIALS

N.A.

7.0 PERFORMANCE CONTROL

N.A.

8.0 PROCEDURE

8.1 Determination of Fuel Core Damage by Hydrogen Production

8.1.1 Obtain sample analysis by grab sampling via the Post Accident Sample Panel if the hydrogen igniters have been energized or there is evidence of hydrogen burn. The $H_2 + O_2$ concentrations are determined per Reference 3.9.

NOTE

Use one Attachment 3 form for Drywell Volume and one for Containment Volume.

8.1.2 Refer to Attachment 3 for calculation using this method.

8.2 Determination of Fuel Core Damage by Iodine Concentration

NOTES

1. During a reactor accident, all of the iodine is assumed to remain in the reactor coolant.

2. If mixing of reactor coolant with water of the suppression pool occurs, the suppression pool may be sampled and the volumes of the reactor coolant loop plus the water volume of the suppression pool have to be used to calculate the total amount of I-131 release from the core.

3. If the accident is such that no mixing with suppression pool water occurs, the reactor coolant loop must be sampled and only its water volume be used in the calculation.

8.2.1 Obtain a liquid sample per Reference 3.2 and perform the analysis per reference 3.11 to determine the I-131 concentration.

8.2.2 Convert the measured specific activity to the total I-131 activity released by using the appropriate water volume.

8.2.3 Obtain correction factors from Attachment 1 and 2 for X and Y to normalize the data for comparison.

8.2.4 Calculate the percent of core damage by dividing the corrected released I-131 activity by the total I-131 activity of the reactor core.

8.2.5 Refer to Attachment 4 for calculation using this method.

8.3 Determination of Fuel Core Damage by Cs-137

NOTES

1. During a reactor accident, all of the Cesium is assumed to remain in the reactor coolant.

2. If mixing of reactor coolant with water of the suppression pool occurs, the suppression pool may be sampled and the volumes of the reactor coolant loop plus the water volume of the suppression pool have to be used to calculate the total amount of Cesium release from the core.

3. If the accident if such that no mixing with suppression pool water occurs, the reactor coolant loop must be sampled and only its water volume be used in the calculation.

8.3.1 Obtain a liquid sample per Reference 3.2 and perform the analysis per Reference 3.11 to determine the Cs-137 concentrations.

8.3.2 Convert the measured specific activity to a total Cs-137 activity released by using the appropriate water volume.

8.3.3 Obtain correction factors from Attachment 1 and 2 for X and Z to normalize the data for comparison.

8.3.4 Calculate the percent of core damage by dividing the corrected released Cs-137 activity by the total available Cs-137 activity of the reactor core.

8.3.5 Refer to Attachment 5 for calculation using this method.

8.4 Determination of Fuel Core Damage by Xe-133

8.4.1 Obtain a gaseous sample per Reference 3.3 and perform the analysis per reference 3.13 to determine Xe-133 concentration.

8.4.2 Obtain the Power Correction Factor (Y) from Attachment 1.

8.4.3 Calculate the percent of core damage by dividing the corrected released Xe-133 activity by the total available Xe-133 activity of the reactor core.

8.4.4 Refer to Attachment 6 for calculation using this method.

8.5 Determination of Fuel Core Damage by Kr-85

8.5.1 Obtain a gaseous sample per Reference 3.3 and perform the analysis per Reference 3.13 to determine (Kr-85) concentration.

8.5.2 Obtain the Power Correction Factor Z from Attachment 1.

8.5.3 Calculate the percent of core damage by dividing the corrected released Kr-85 activity by the total available Kr-85 activity in the reactor core.

8.5.4 Refer to Attachment 7 for calculation using this method.

8.6 Estimation of Core Damage

8.6.1 Enter the results from analysis of methods 8.1, 8.2, 8.3, 8.4 and 8.5 on Attachment 8 and calculate the average estimated cladding damage.

8.6.2 Sign Attachment 8 and include all other attachments completed.

8.6.3 Submit to the Chemistry/Core Damage Assessment Coordinator for verification.

8.6.4 Evaluate the results of Attachment 8 and notify the Technical Support Center - Emergency Director as per Reference 3.12 of the Core Status Estimation.

8.7 Estimation of Fuel Core Damage from Standardized Graphs

8.7.1 Calculate the corrected released I-131 activity (Item 3. of Attachment 4) as per Steps 8.2.1 to 8.2.2.

8.7.2 Divide by the primary coolant mass and multiply by the primary coolant mass correction factor (Section 4.1)

8.7.3 For the standardized specific I-131 activity of Step 8.7.2 obtain the corresponding value for the estimated fuel core damage from Attachment 9.

8.7.4 Record the result on the appropriately marked Attachment 8 form.

8.7.5 Calculate the corrected released Cs-137 activity as per Steps 8.3.1 to 8.3.3 (Item 2. of Attachment 5)

8.7.6 Repeat Stpes 8.7.2 to 8.7.4 for Cs-137 by using Attachment 10 in Step 8.7.3.

8.7.7 Calculate the corrected released Xe-133 activity as per Subsection 8.4 (Item 2 on Attachment 6).

8.7.8 Divide by the containment gas volume and multiply by the containment gas volume correction factor (Section 4.1).

8.7.9 For the standardized Xe-133 activity concentration of Step 8.7.8 obtain the corresponding value for the estimated fuel core damage from Attachment 11.

8.7.10 Record the result on the appropriately marked Attachment 8 form.

8.7.11 Calculate the corrected released Kr-85 activity as per Subsection 8.5 (Item 2 on Attachment 7)

8.7.12 Repeat Steps 8.7.8 to 8.7.10 for Kr-85 by using Attachment 11 in Step 8.7.9.

8.7.13 Obtain and record on Attachment 8 the value of the fuel core damage derived from the containment H_2 concentration of the Graph Attachment 13 using the method outlined on Attachment 14.

8.7.14 Calculate the average Estimated Cladding Damage from the four entries for I-131, Cs-137, Xe-133, Kr-85 and H_2.

8.7.15 Proceed as per Steps 8.6.2 to 8.6.4.

9.0 ACCEPTANCE CRITERIA

N.A.

"END"

1. **Correction factor for I-131 and Xe-133 (Y)**

To correct for core isotopic inventory if fuel damage is suspected, calculate the correction factor for I-131 and Xe-133 by using the following equation:

$$Y = \frac{100\%}{\text{Average thermal power over the last 30 days in percent}}$$

2. **Correction factor for Cs-137 and Kr-85 (Z)**

For calculations employing Cs-137 and Kr-85 use the following equation for the correction factor Z:

$$Z = \frac{100\%}{\text{Average thermal power over the last 3 years in percent}}$$

Refer to Reference 3.8 if more accurate Power Correction is desired

3. **Reference Plant Parameters (Reference 3.8)**

Rated Thermal Power: 3579 MW_t

# of Fuel Bundles: 748

Total primary coolant mass: 3.92E9 g
(reactor + suppression pool)

Total containment and drywell
gas space volume 4.0 E10 cc

4. **Primary Coolant Mass Correction Factor (F_w)**

$$F_w = \frac{\text{RBS Coolant Mass}}{\text{Ref. Plant Coolant Mass}} = \frac{3\text{ E9 g}}{3.92\text{E9g}} = \underline{\qquad}$$

5. **Containment Gas Volume Correction Factor (F_g)**

$$F_g = \frac{\text{RBS Containment \& Drywell Gas Vol.}}{\text{Ref. Plant Containment Gas Vol.}} = \frac{3.87\text{ E10 cc}}{4.00\text{E10 cc}} = \underline{0.967}$$

Normal Reactor Coolant System sample temperature is approximately 90°F. Determine the appropriate Reactor Coolant temperature at the time of sampling and select the associated density correction factor X from the table.

Reactor Coolant Sample Temperature °F

REACTOR COOLANT TEMPERATURE	DENSITY CORRECTION FACTOR (X)
100	.998
150	.985
200	.968
250	.947
300	.923
350	.895
400	.864
450	.825
500	.788
550	.740
560	.729
570	.718
580	.708
590	.694
600	.681

1. Volume Considered

[ ] Containment (Volume = 1.12 E6 ft^3)

[ ] Drywell (Volume = 2.47 E5 ft^3)

2. H_2 Volume at Accident Conditions

a. From H_2 Data

(____% H_2) (____ ft^3 Vol.)/100% = ____________ ft^3 H_2

b. From O_2 Depletion

(____% O_2 Normal) - (___% O_2 Post Acid) = ____________% O_2 Depl.

(____% O_2 Depl)x(2)x(____ ft^3 Vol)/100% = ____________ ft^3 H_2 (O_2)

c. Total Volume of Lib. H_2 in Vol. (Add a. and b.)

(____ ft^3 H_2) + (____ ft^3 H_2 (O_2)) = ____________ ft^3 H_2 Tot.

3. H_2 Volume at Standard Pressure and Temperature (scf)

T = _________ °C (In considered volume)

P = _________ psia (In considered volume)

(____ ft^3 H_2 Tot.) (1 + ___ °C/273°C) (____ psia/14.7 psia)

= _____ scf H_2 in [ ] Containment

[ ] Drywell

4. Total Mass of Zirconium Reacted

______ scf H_2 Cont. + ______ scf H_2 Dryw. = ____________ scf H_2 Tot.

______ scf H_2/(8.0 scf H_2/lb Zr React.) = ____________ lb Zr

5. Percentage of Core Cladding

100% x (______ lb Zr React)/((LATER) lb Zr in core)

= _________% of Total Zr Reacted

1. Total Activity of I-131

NOTE

Determine the transient involved to account for the total iodine activity. Make the necessary corrections as related to the specific incident.

a. Measured in the Suppression Pool + Reactor Coolant (N/A if suppression pool not used)

(______ uCi/ml)(LATER ml)(1E-6) = __________ Ci (SP & RC)

b. Measured in the Reactor Coolant

(______ uCi/ml)(LATER ml)(1E-6) = __________ Ci (RC)
(N/A if suppression pool used)

2. Decay Calculation to Time of Reactor Shutdown T_o

Activity @ counting time T x $e^{+0.693(T - T_o)/T1/2}$ = Activity released at T_o

(____ Ci) · $e^{+0.693 (\_\_\_\_ h)/193.2h}$ = __________ Ci @ T_o

3. Power and Density Correction

(I-131 Activity from 2.)(Y)(X) = corrected Activity released at T_c

(______ Ci) (________) (________) = __________ Ci

(Y = Average Capacity Factor for previous 30 days)

4. Percent of Core Damage

100% x (________________ Ci released/(LATER) Ci available) = ________%

1. Total Activity of Cs-137

   a. Measured in the Suppression Pool (N/A if suppression pool not used)

   (_______ uCi/ml)(LATER ml)(1E-6) = __________ Ci (SP & RC)

   b. Measured in the Reactor Coolant (N/A if suppression pool used)

   (_______ uCi/ml)(LATER ml)(1E-6) = __________ Ci (RC)

2. Power and Density Correction

   (Activity Cs-137 from a. or b.) (X)(Z) = Activity released at T_o

   (_______ Ci) (__________) (__________) = __________ Ci released @ T_o

   (Z = Average Capacity Factor for Previous 3 Years)

3. Percent of Available Cesium 137 Released

   100% x (____________ Ci released/)(LATER) Ci available = __________%

1. Total released Xe-133 Activity

   a. Measured in the Containment

   (______ uCi/cc)(3.87E10 cc)(1E-6) = _____ Ci (Drywell and Cont)

   Assumes approximate equal distribution in drywell and containment

   b. Measured in the drywell

   (______ uCi/cc)(6.99E9 cc) (1E-6) = _____ Ci (Drywell)

   c. Xenon 133 Measured in the containment

   (______ uCi/cc) (3.17 E10 cc) (1E-6) = __________ Ci (Cont.)

NOTE

Use either a. or the sum of b. and c. to fit the specific situation.

2. Decay Evaluation to Time of Reactor Shutdown

   (Activity @ time T of count) $e^{+0.693 (T-T_o)/T1/2}$ = Activity rel. @ T_o

   (Ci @ T)$e^{+0.693 (\_\_h)/126.5h}$ = ____________ Ci released @ T_o

3. Power Correction

   (Activity Xe-133 from 2.)(Y) = Corrected Activity @ T_o

   (Ci) ( ) = __________ Ci @ T_o

4. Percent of Available Xe-133 Released

   100% x (______________ Ci released)/(LATER) Available = ____________ %

1. Total Released Kr-85 Activity

a. Measured in the Containment

(_____ uCi/cc)(3.87E10 cc)(1E-6) = _______ Ci (Drywell & Cont)

Assumes approximate equal distribution in drywell and containment

b. Measured in the Drywell

(_____ uCi/cc)(6.99E9 cc) (1E-6) = _______ Ci (Drywell)

c. Measured in the Containment

(_____ uCi/cc) (3.17E10 cc) (1E-6) = _______ Ci (Cont)

NOTE: Use either a., b., or c. or the sum of b. and c. to fit the specific situation of the Kr-85 distribution present.

2. Power Correction

(Ci Kr-85 from above) (Z) = Corrected Activity @ T_0

(____________ Ci) (____________) = _________ Ci @ T_0

(Z = Average Capacity Factor for Previous 3 Years)

3. Percent of Available Kr-85 Released

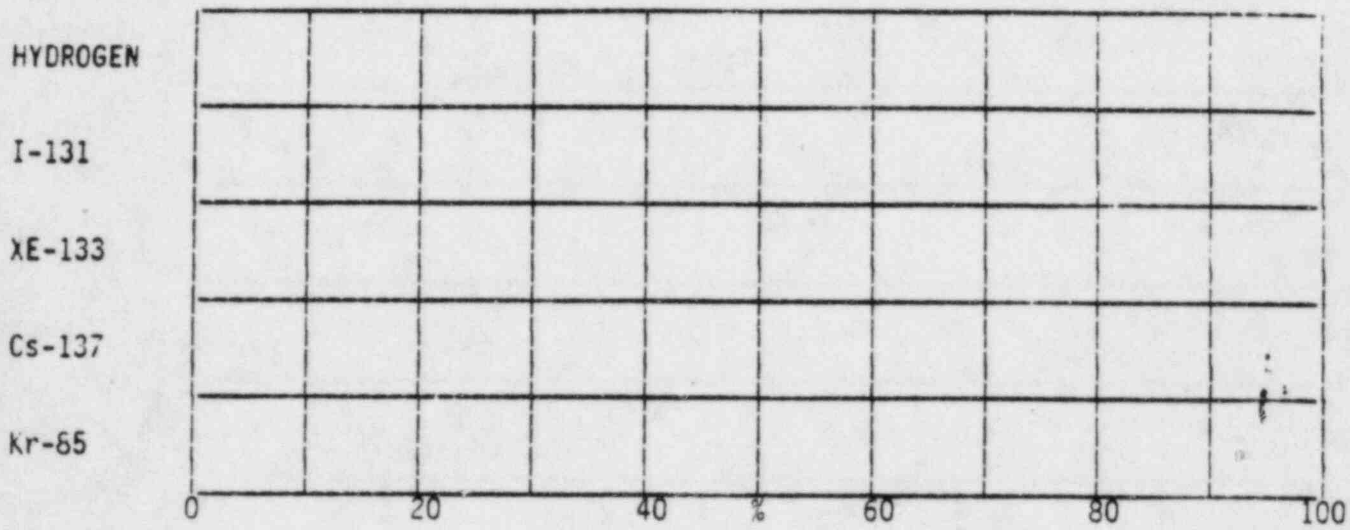
100% x (____________ Ci released)/(LATER) Ci Available = ______________ %

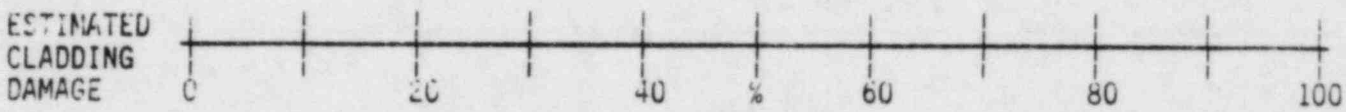
ATTACHMENT - 8	CORE DAMAGE GRAPH

METHOD: CALCULATION (8.1 to 8.6) ☐

GRAPHS (8.7) ☐

% CORE DAMAGE





Performed by ______________________________ ______________
Signature Date

Reviewed by ______________________________ ______________
Chemistry Core Damage Assessment Coordination Date

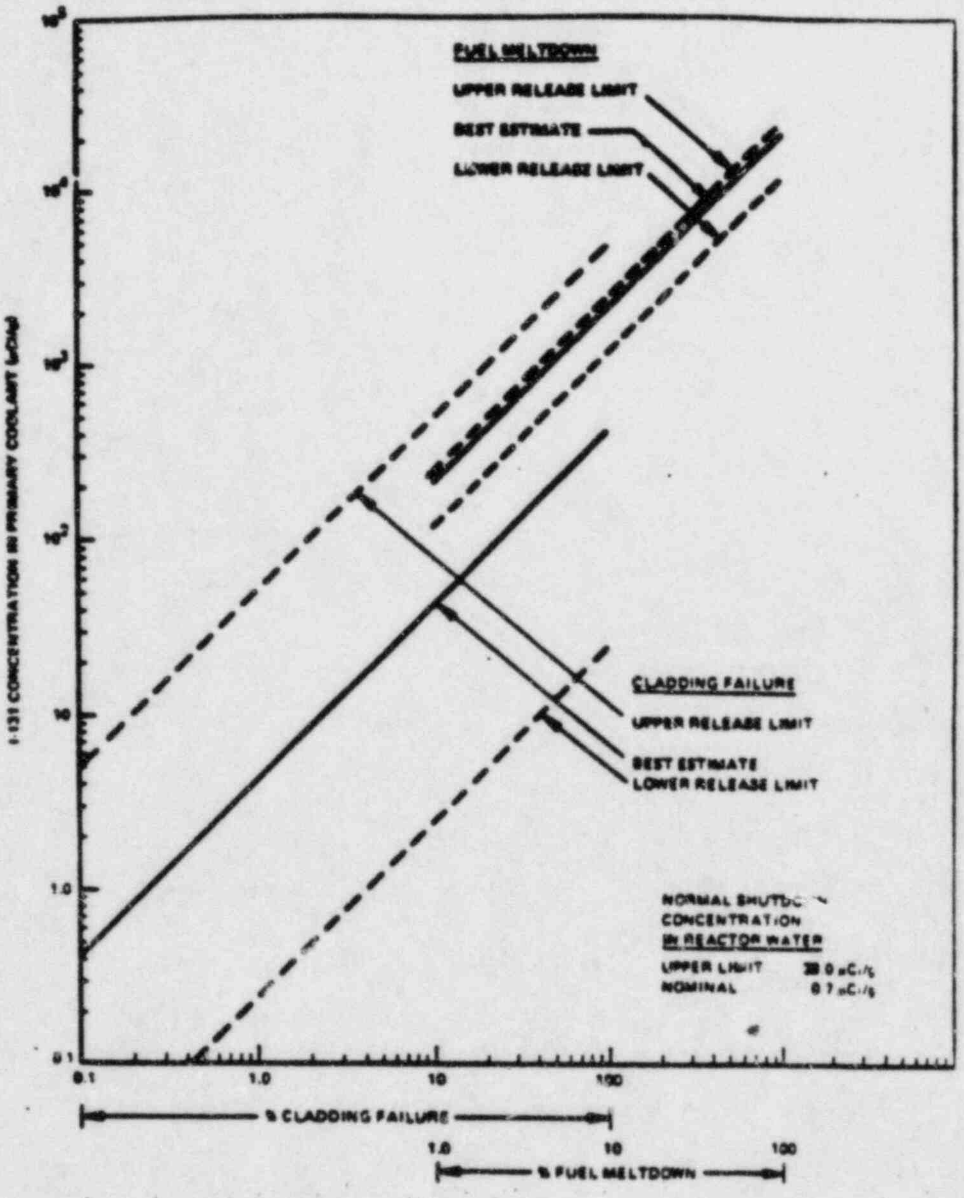


Figure 1. Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

2-4

RPC-000.067

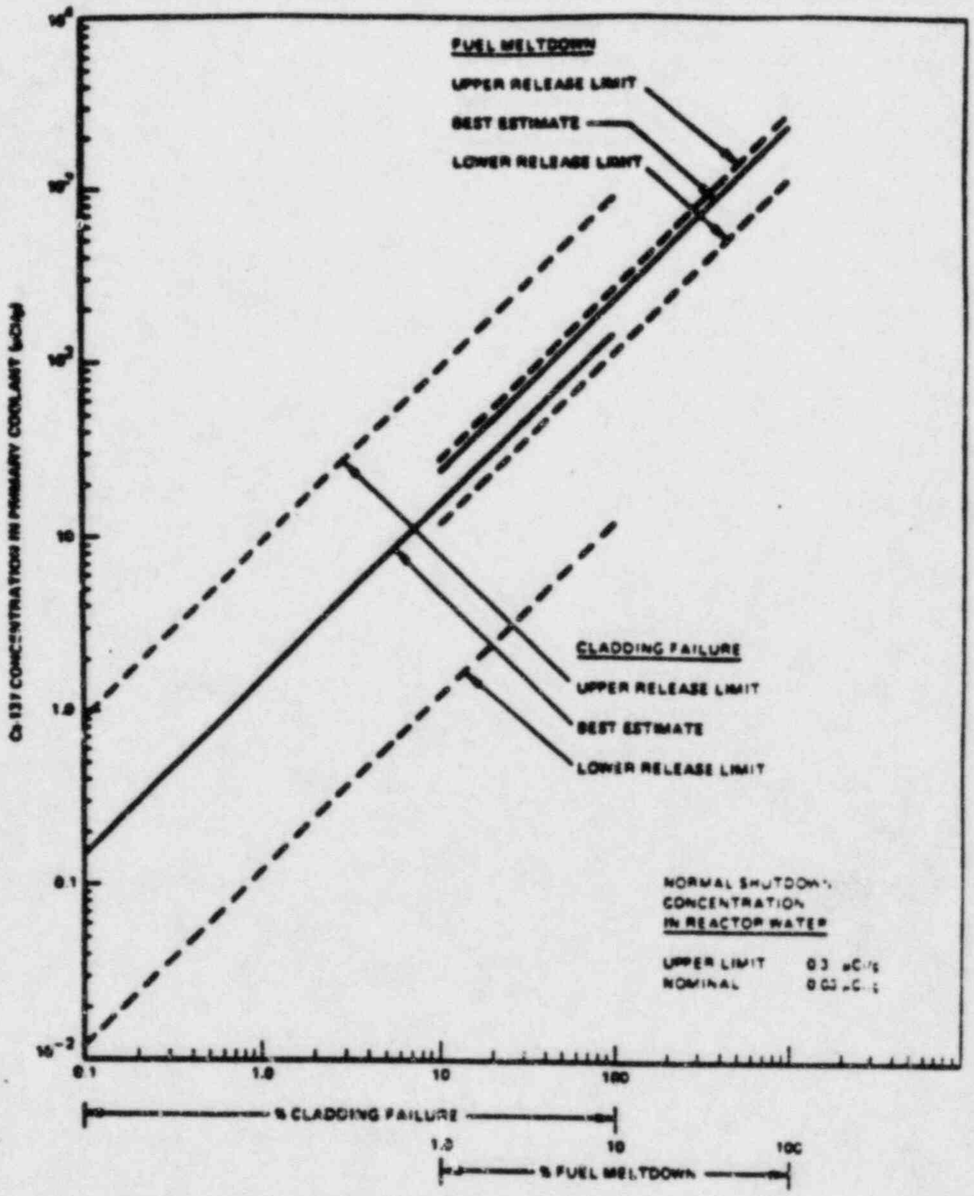


Figure 2. Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

2-5

RPC-000.068

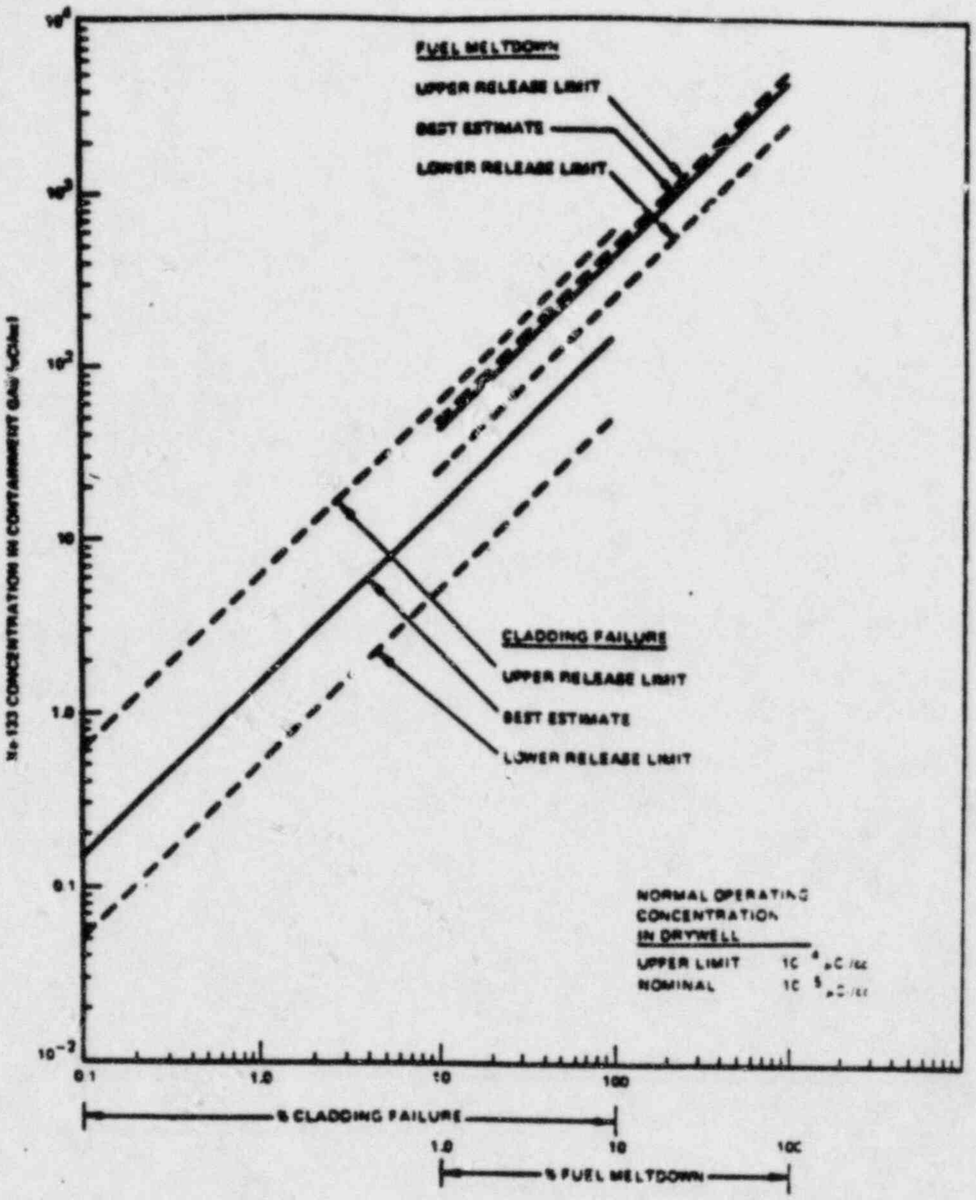


Figure 3. Relationship Between Xe-133 Concentration in the Containment Gas (Drywell + Torus Gas) and the Extent of Core Damage in Reference Plant

2-6

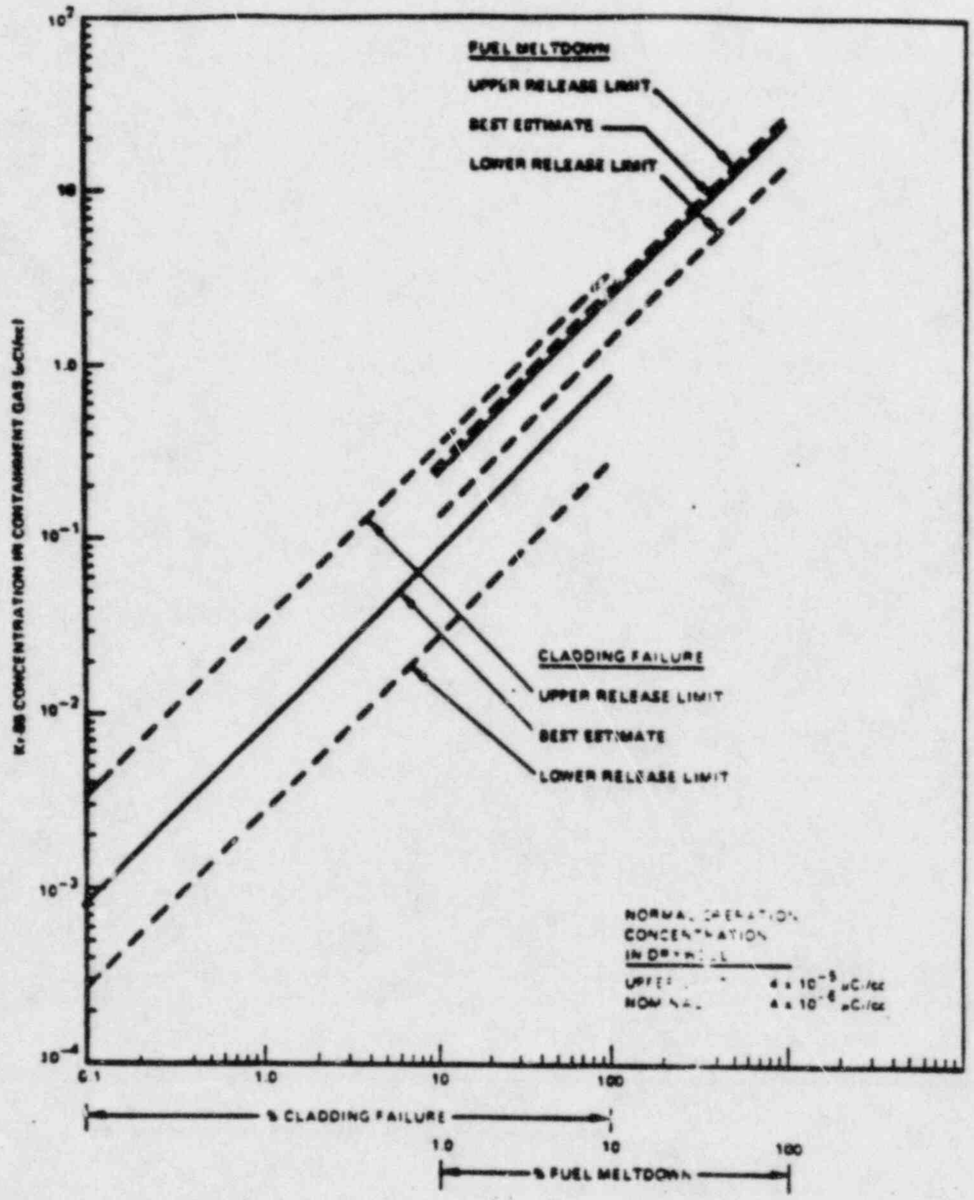


Figure 4. Relationship Between Kr-85 Concentration in the Containment Gas (Drywell - Torus Gas) and the Extent of Core Damage in Reference Plant

?-7

1. Obtain the total scf H_2 volume released into the containment and the dry well (Item 3 of Attachment 3) as per Subsection 8.1.

2. Calculate the % H_2:

$$\% H_2 = \frac{\text{scf } H_2}{\text{Cont. Vol}} \times 100\% = \frac{(\qquad\qquad)}{1.12 \text{ E6 ft}^3} = \_\_\_\_\_\_\%$$

3. For the resultant % H_2 value obtain the corresponding value for the % Metal-Water Reaction (%MW) for the 748 (MK III) Reference Plant (right vertical axis).

4. Calculate the % MW for RBS:

$$\% MW = (\%MW_{ref}) \frac{748}{N} \left(\frac{V}{1.36 \text{ EG}}\right)$$

$$\times 0.987 = \_\_\_\_\_\_\_\_\%$$

RBS Number of Fuel Bundles = 624

= RBS Containment Net Air Volume = 1.12 E6 ft^3

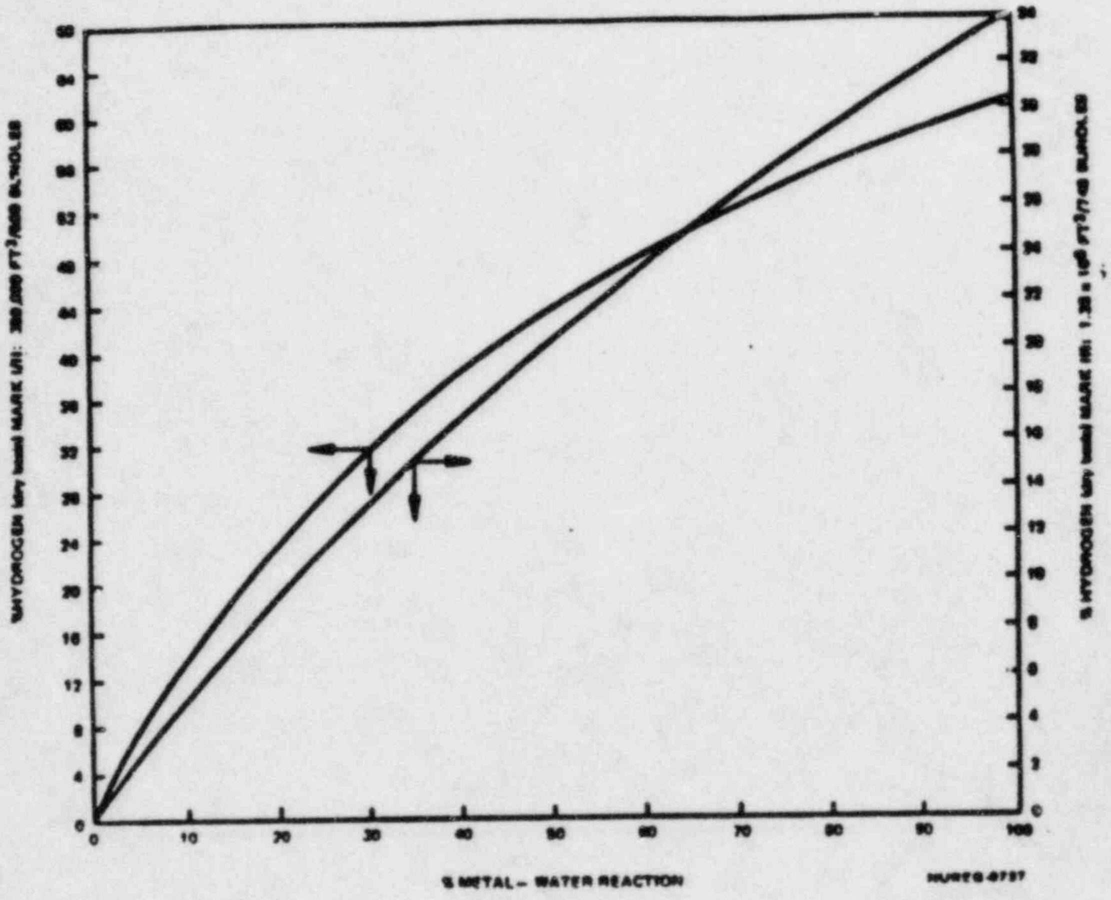


Figure A-1. Hydrogen Concentration for Mark I/II and III Containments as a Function of Metal-Water Reaction