PROCEDURE COVER SHEET

PENNSYLVANIA POWER & LIGHT CO. SUSQUEHANNA STEAM ELECTRIC STATION	
ESTIMATION OF CORE DAMAGE DURING AN EMERGENCY	EP-IP-048 Revision 1 Page 1 of 27

Effective Date 11-21-84 Expiration Date 11-21-86

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PROCEDURE TYPE: PORC X, NON-PORC ____, Alternate Review ____

PORC MTG. NO. 84-244 (If applicable)

Prepared by Connie Kuchwa Date 11/20/84

Reviewed by Michael B. Detamore Date 11/20/84

Recommended: [signature] Date 11/20/84
Section Head/Manager

[signature] Date 11-20-84
Superintendent of Plant

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ATTACHMENTS

ATTACHMENTS (Cont'd)

1.0 PURPOSE

The purpose of this procedure is to determine the degree of reactor core damage from the measured fission product concentrations in either the water or gas samples taken from the primary system under accident conditions.

2.0 SCOPE

The procedure involves calculations of fission product inventories in the core and the release of inventories into the primary system under postulated loss-of-coolant accident (LOCA) conditions. The fuel gas fission products are assumed to be released upon the rupture of fuel cladd~~ing,~~ and the majority of fission product inventories in the core will be released when the fuel is melted at higher temperatures.

3.0 REFERENCES

3.1 Lin, Chien C.: PROCEDURE FOR THE DETERMINATION OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS. General Electric Co.,: August, 1982

3.2 EP-IP-034, Reactor Building Sample Station Emergency Sampling

3.3 EP-IP-042, Chemistry Lab Emergency Preparation

3.4 EP-IP-043, PASS Small Volume Liquid Sample

3.5 EP-IP-045, PASS 10-ml Gas Sample

3.6 EP-IP-046, PASS Iodine/Particulate Sample

3.7 EO-00-025-0, RPV Flooding

3.8 SSES Technical Specifications

4.0 RESPONSIBILITIES

4.1 It is the responsibility of the TSC COORDINATOR upon the activation of the PASS station to:

4.1.1 Obtain sample data on:

a. I-131 activity in the reactor coolant and/or

b. Xe-133 activity in the primary containment in accordance with procedures EP-IP-045 and EP-IP-046.

4.1.2 Use the above sample data to compute an initial estimate of core degradation as described in Section 6.0 of this procedure.

4.1.3 Communicate this initial estimate of core degradation to the Emergency Director.

4.2 Upon the activation of the EOF, it is the responsibility of the TECHNICAL SUPPORT MANAGER to:

4.2.1 Provide the initial estimate of core damage to the ENGINEERING SUPPORT LEADER.

4.2.2 Initiate subsequent PASS measurements of fission product activities, including I-131 and/or Xe-133, on an as needed basis and provide this data to the ENGINEERING SUPPORT LEADER.

4.3 It is the responsibility of ENGINEERING SUPPORT LEADER to:

4.3.1 Use the above fission product data in conjunction with corroborative plant data and phenomena to re-estimate core damage, and inform the Technical Support Manager of the revised estimate.

4.3.2 Upon subsequent receipt of relevant plant data, accommodate the new information in his evaluation of core damage, and inform the Technical Support Manager of the revised estimate.

5.0 DEFINITIONS

5.1 Clad Failure - The ability to contain fission products in the gap is lost, but the cladding structural integrity is maintained.

5.2 Fuel Melt - the U02 pellets are or have been in a molten state.

5.3 Fuel Overheat - The cladding structural integrity is lost, but no fuel melting has occurred.

5.4 PASS - Post-Accident Sampling System

6.0 INSTRUCTIONS

The TSC COORDINATOR shall perform the following with assistance and re-evaluation from the TECHNICAL SUPPORT MANAGER and the ENGINEERING SUPPORT LEADER:

6.1 Evaluate the following conditions from the Tech Specs:

6.1.1	Off gas activity	3.11.2.7	Radioactive Effluents Main Condenser
6.1.2	I-131 Equivalent	3/4.4.5	Specific Activity
6.1.3	Fission Product $\bar{E}$	3/4.4.5	Specific Activity

6.2 The estimate of core damage shall be made using PASS samples and verified using reactor water level, containment hydrogen concentration and the presence of less volatile fission products.

6.3 Estimate of Core Damage based on PASS Samples.

6.3.1 Determination of the optimum PASS Sample Point

a. Gas Samples

The optimum gas sample point is a function of the event type. Table #1 lists the sample points as a function of the event.

b. Liquid Samples

The optimum sample point for all events is at the jet pumps. If a non-break or small break event has occurred and the power level is less than 1%, the water level should be raised to the moisture separators. This induces natural circulation and ensures a representative liquid sample is obtained. If the reactor pressure is insufficient to allow samples to be obtained from the jet pumps, then the RHR sample point should be used.

6.3.2 Conversion of Sample Activities to Core Damage Estimates

a. Record the time the reactor tripped in Form EP-IP-048-1.

b. Obtain samples of I-131, Cs-137, Xe-133, and Kr-85 consistent with step 6.3.1 of this procedure and with EP-IP-034, EP-IP-042, EP-IP-043, EP-IP-045, and EP-IP-046.

Record the time the sample specific activities are determined, the containment pressure and temperature, the sample temperature and pressure, and the isotopic concentrations in Form EP-IP-048-1.

Determine the normalized concentrations of I-131, Cs-137, Kr-85 and Xe-133 using the following equations:

$$Cwi = \frac{RWCi\ \mu ci/gm \times 2.9\times10^{8} gm + PWCI\ \mu ci/gm \times 3.6\times10^{9} gm}{3.89 \times 10^{9} gm}$$

$$Cgi = \frac{DWCi\ \mu ci/cc \times 6.79\times10^{9} cc + WWci\ \mu ci/cc \times 4.36\times10^{9} cc}{1.12\times10^{10} cc}$$

Cwi = Concentration of isotope i in the primary coolant
Cgi = Concentration of isotope i in the containment gas
RWCi = Concentration of isotope i from PASS reactor water sample
PWCi = Concentration of isotope i from PASS suppression pool water sample
DWci = Concentration of isotope i from PASS drywell gas sample
WWCi = Concentration of isotope i from PASS wetwell air space sample

Record the normalized isotopic concentrations in Row 1 Form EP-IP-048-2.

NOTE: Accurate measurement of Cs-137 and Kr-85 may be difficult to obtain until the shorter lived isotopes have decayed away a few weeks after shutdown.

d. Correction for Temperature and Pressure

Correct the measured gaseous specific activity using the following relationship.

$$TCF = \frac{(\text{Containment Pressure}) \times (\text{Sample Temperature})}{(\text{Sample Pressure}) \times (\text{Containment Temperature})}$$

Here TCF is the thermodynamic correction factor.

Record the value of TCF in Row 2 of Form EP-IP-048-2.

e. Correction for Power History

The referenced fission product inventories used in this procedure are based on the reactor operating at full power for three years. Since this is not the case, corrections are made for the power history. The fission product inventory correction factor, FIi, is computed

for each isotope using the following equations and Forms EP-IP-048-3 and EP-IP-048-7.

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

Here,

λi - decay constant of isotope (days)
Pj - is the steady state power during period j (MWt)
Tj - is the duration of operating period j (days)
Tj° - is the time between the end of operating period j and the time of last reactor shutdown. (This is the shutdown associated with the event leading to core damage.) (days)

The steady state power variation during an operating period, j, should be limited to ±20%.

For a particular short-lived isotope, i, a calculation for only a period of 6 half-lives of reactor operation time before reactor shutdown should be accurate enough. It should be pointed out that the computer calculation of core inventory takes into account the fuel burnup, plutonium fission and neutron capture reactions. The correction factor calculated from this equation may not be entirely accurate, but the error is insignificant in comparison to the uncertainties in the fission product release fractions and other assumptions.

Enter the appropriate values in Form EP-IP-048-3 and EP-IP-048-7 and record the values of FIi in Form EP-IP-048-2.

f. Corrected Fission Product Concentration

Determine the corrected fission product concentration by multiplying the normalized fission product specific activity by thermodynamic correction factor, the power history correction factor and the plant parameter correction factor as indicated below:

$$C_{iw} = C_{wi}e^{\lambda_i t} \times F_{Ii} \times F_w$$

$$C_{ig} = C_{gi}e^{\lambda_i t} \times F_{Ii} \times F_g \times TCF$$

where t = time between reactor shutdown and the sample time

$$Fw = \frac{3.89 \times 10^9 g}{3.92 \times 10^9 g} = 0.99$$

$$Fg = \frac{5.04 \times 10^9 cc}{4 \times 10^{10} cc} = 0.13$$

Record the values of Ciw and Cig in Row #5 of Form EP-IP-048-2.

g. Estimate of Core Damage Using Corrected Fission Product Concentrations (Ciw and Cig)

Obtain an estimate of core damage by reading off Figure 1 through 4, using the appropriate corrected fission product concentrations listed in Row #5 of Form EP-IP-048-2.

Record these values on FORM EP-IP-048-4.

6.4 Water Level Determination

6.4.1 Determine from the shift supervisor if water level has dropped below TAF. Enter the elapsed time after shutdown, the core became uncovered and amount of time the core was uncovered in FORM EP-IP-048-4.

6.4.2 Determine the maximum core uncovery time using entry 2.2 of FORM EP-IP-048-4 and Figure 5. Record this value in FORM EP-IP-048-4.

6.4.3 If entry 2.2 exceeds entry 2.3, the maximum core uncovery time has been exceeded.

6.5 Integration of Containment Hydrogen

6.5.1 Obtain the containment hydrogen monitor reading in mole %, (H). Record this value in FORM EP-IP-048-4. Estimate the amount of metal water reaction using (H) and Figure 7. Record the % of clad that has undergone water metal reaction in FORM EP-IP-048-4.

6.5.2 Correct the percent of metal water reaction for the reference plant to SSES specific numbers using the following relationship:

%MW = 0.736 x %MW

Record %MW in FORM EP-IP-048-4.

6.6 Integration of Fission Product Release Fractions into the Core Damage Estimate.

6.6.1 Take a coolant sample from the appropriate location as discussed in Section 6.3.1.

6.6.2 Determine if elements from the Tellurium, Noble Metal, Alkaline Earth, Rare Earth and Refractory groups are present in the coolant sample. These elements are identified in FORM EP-IP-048-5.

6.6.3 If these elements are not identified in the coolant sample then core melt is not indicated and this fact should be indicated in FORM EP-IP-048-4. If these elements are identified, enter their concentration in FORM EP-IP-048-5 and make gross estimates of melting as outlined in section 6.6.4.

6.6.4 Estimate of Core Melt Using Less Volatile Fission Products.

a. Compute the activity for the isotope listed in FORM EP-IP-048-5 in the reactor and suppression pool water using the following equation:

$$\text{Total curies} = \left(\begin{array}{c}\text{Reactor water}\\ \text{concentration}\\ \text{(ci/gm)}\end{array} \times 2.9\times10^{8}\ \text{gm}\right) + \left(\begin{array}{c}\text{Suppression}\\ \text{Pool concentration}\\ \text{(ci/gm)}\end{array} \times 3.6\times10^{9}\ \text{gm}\right)$$

Record these values in Column 4 of FORM EP-IP-048-5.

b. Compute the fission product adjustment factor to correct the reference yields to the specific case under consideration as shown in FORM EP-IP-048-3, EP-IP-048-6, and EP-IP-048-7. Enter this adjustment factor on FORM EP-IP-048-5. The definitions of the parameters Pj, Tj and Tj° are located in section 6.3.2.e..

EP-IP-048-5. The definitions of the parameters Pj, Tj and Tj° are located in section 6.3.2.e.

c. Compute the fission products available for release by multiplying the reference fission products available for release by the adjustment factor and enter the results in Columns 10-12 of FORM EP-IP-048-5.

6.6.5 Obtain the estimates of fuel melt by dividing column 5 by column 10-12 and record them in Columns 13-15 of FORM EP-IP-048-5.

6.7 Determining the NRC Core Damage Index

6.7.1 Apply the responses to items 1.5, 2.4, 3.4, 4.1 on FORM EP-IP-048-5 to the logic diagram in Figure 7. The degree the degradation shall be determined by comparing the sample results with the range identified on FORM EP-IP-048-8.

6.7.2 Complete FORM EP-IP-048-8 and forward it and FORM EP-IP-048-4 to the TSC and EOF (if appropriate).

TABLE #1

PASS SAMPLE LOCATIONS FOR GAS SAMPLES

EVENT TYPE	SAMPLE LOCATION
NON-BREAKS	SUPPRESSION POOL ATMOSPHERE
SMALL BREAKS PRIOR TO DEPRESSURIZATION	DRYWELL
SMALL BREAKS AFTER DEPRESSURIZATION	SUPPRESSION POOL ATMOSPHERE
LARGE LIQUID OR STEAM BREAKS IN THE CONTAINMENT	DRYWELL
LARGE LIQUID OR STEAM BREAKS OUTSIDE THE CONTAINMENT	SUPPRESSION POOL ATMOSPHERE

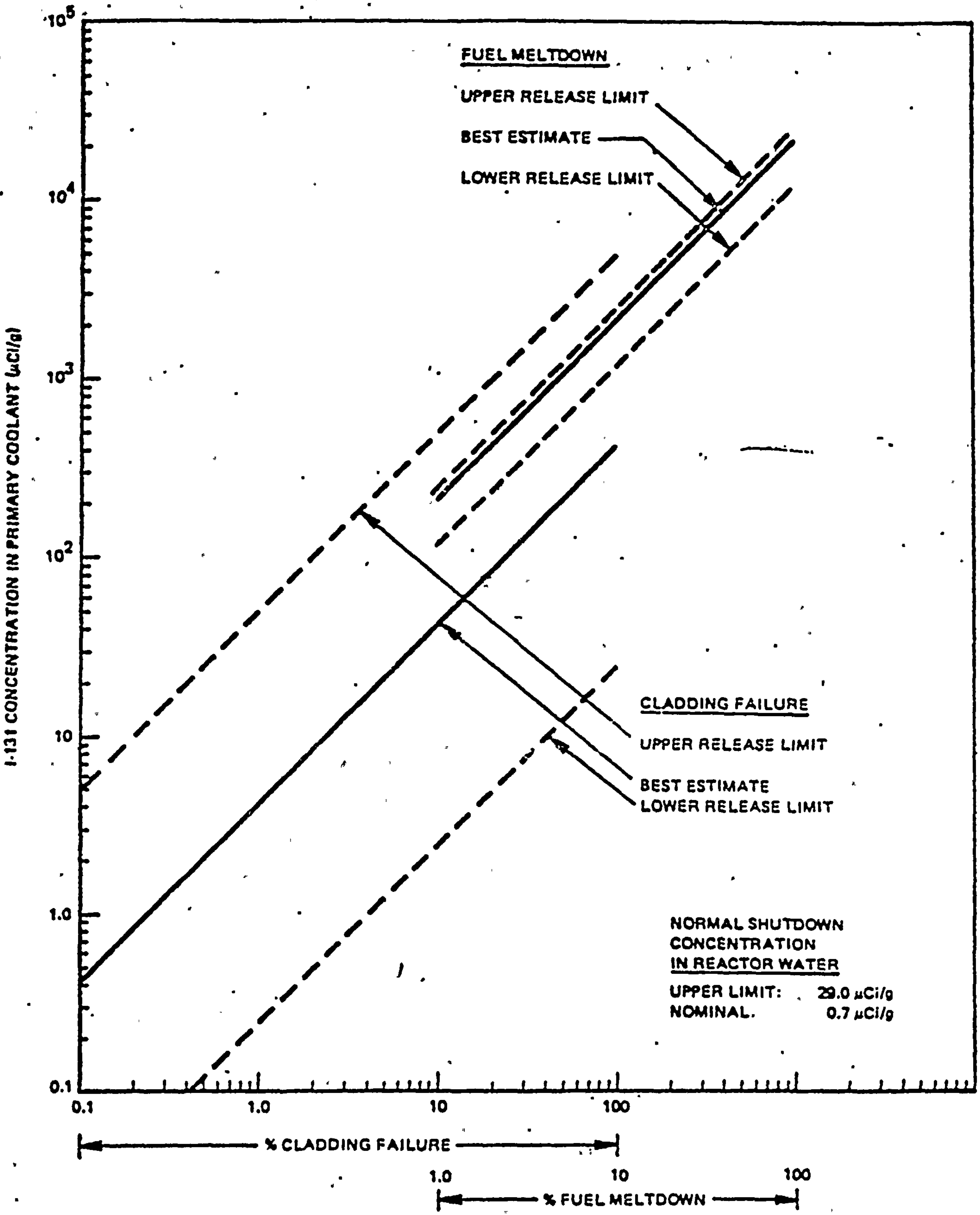


Figure 1 - Relationship Between I-131 Concentration in the Primary Coolant and the Extent of Core Damage in the Reference Plant

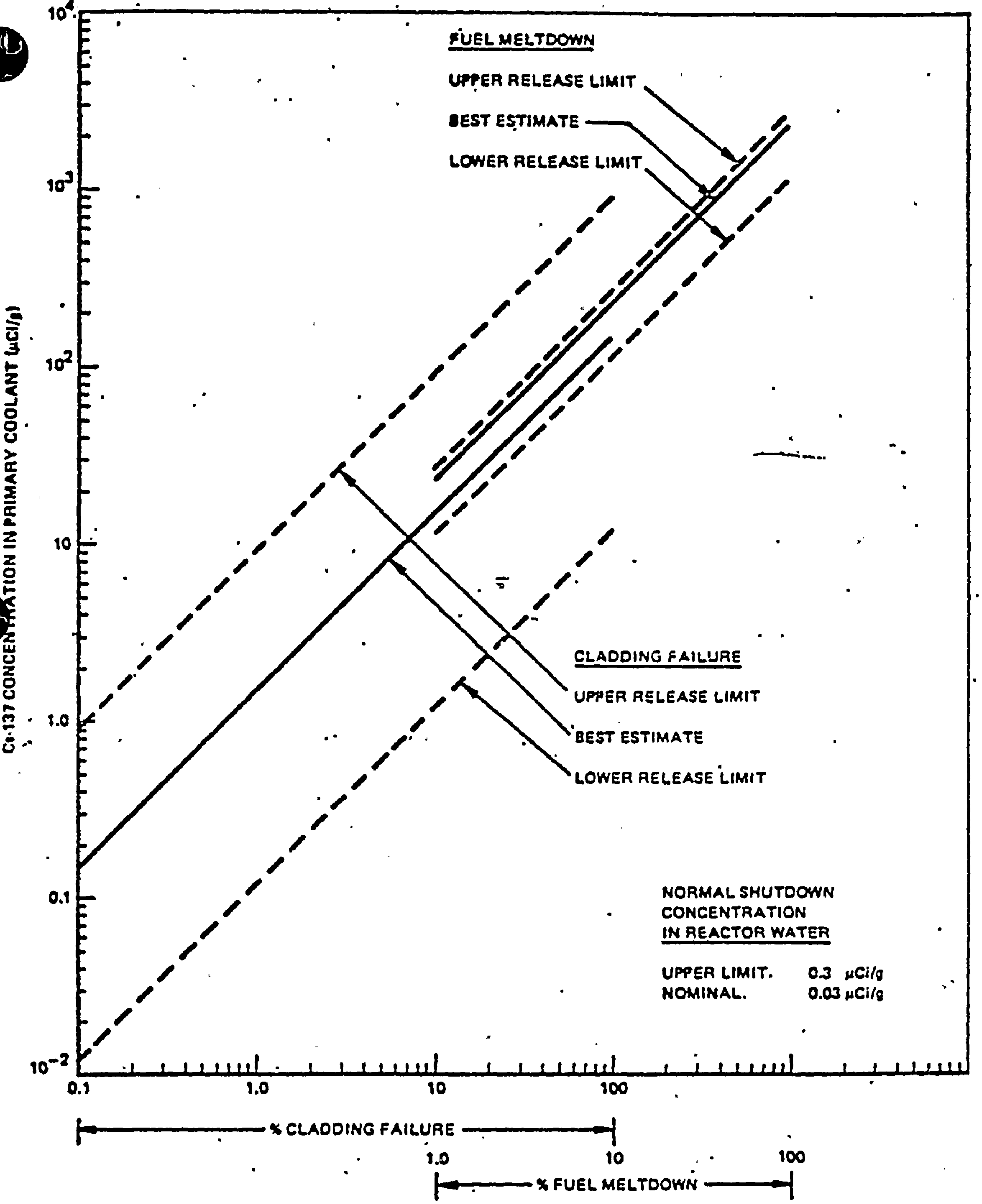


Figure 2 - Relationship Between Cs-137 Concentration in the Primary Coolant and the Extent of Core Damage in the Reference Plant

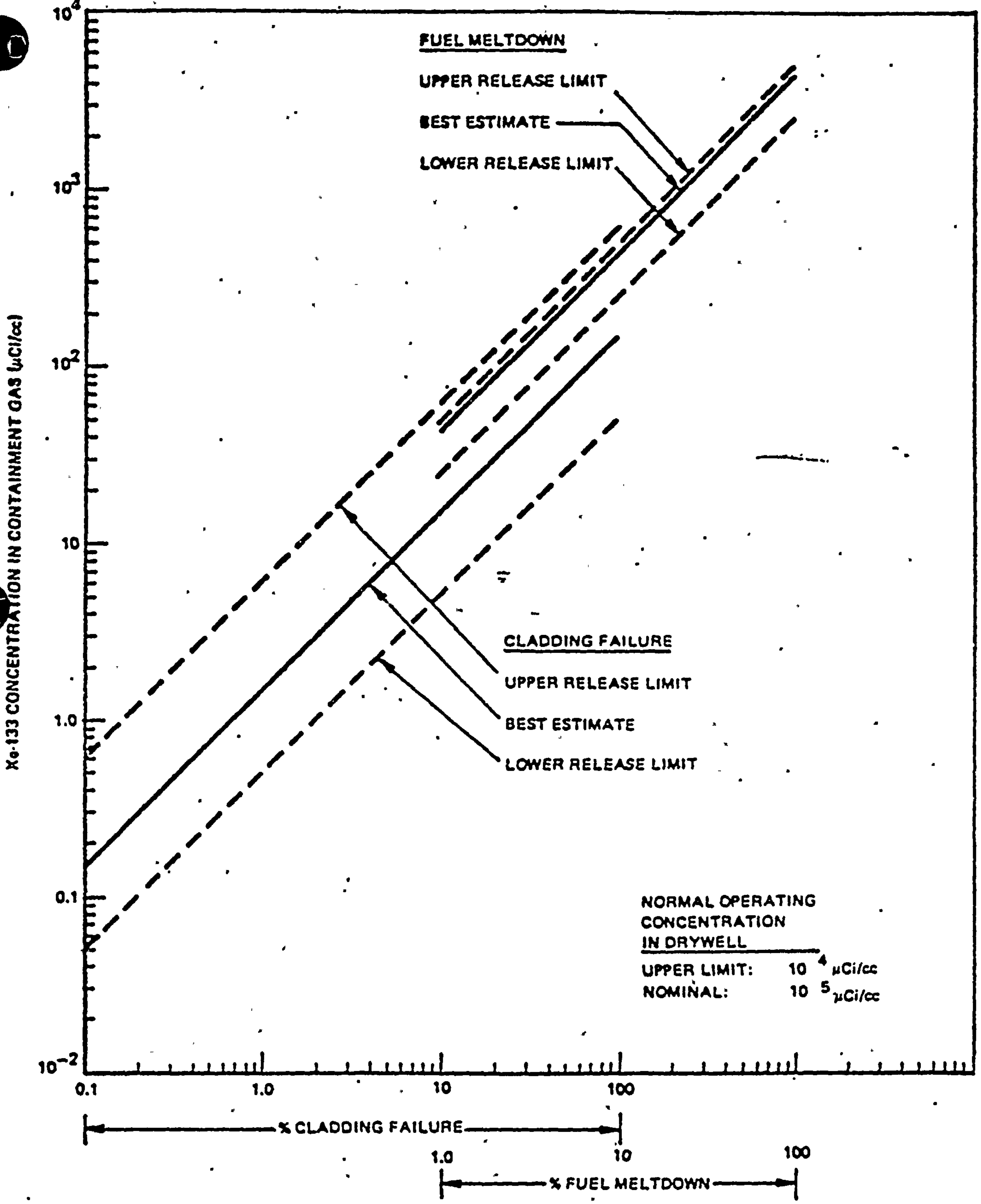


Figure 3 — Relationship Between Xe-133 Concentration in the Primary Coolant and the Extent of Core Damage in the Reference Plant

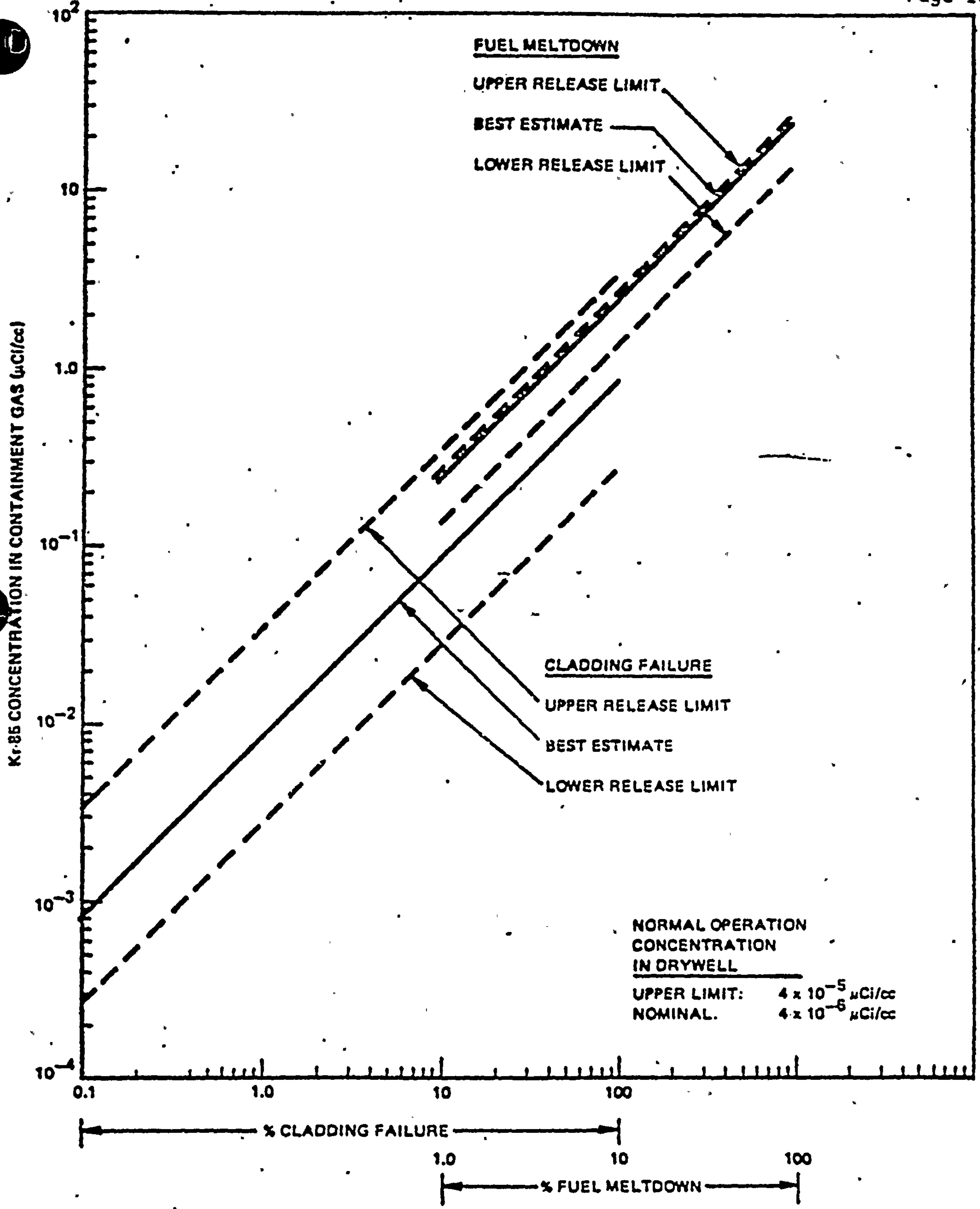


Figure 4 — Relationship Between Kr-85 Concentration in the Primary Coolant and the Extent of Core Damage in the Reference Plant

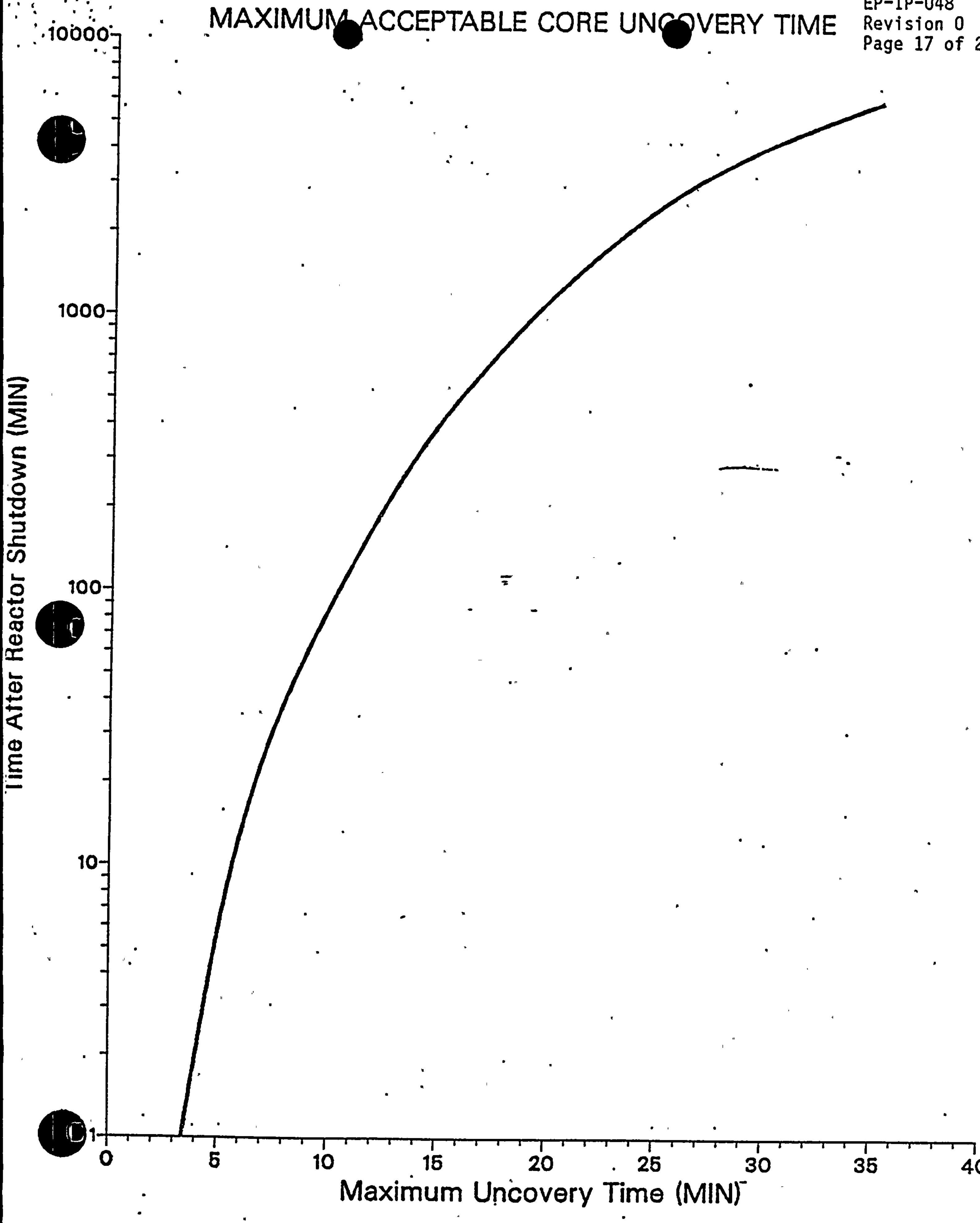

MAXIMUM ACCEPTABLE CORE UNCOVERY TIME
10000
1000
100
10
1
Time After Reactor Shutdown (MIN)
0
5
10
15
20
25
30
35
40
Maximum Uncovery Time (MIN)

Figure 5

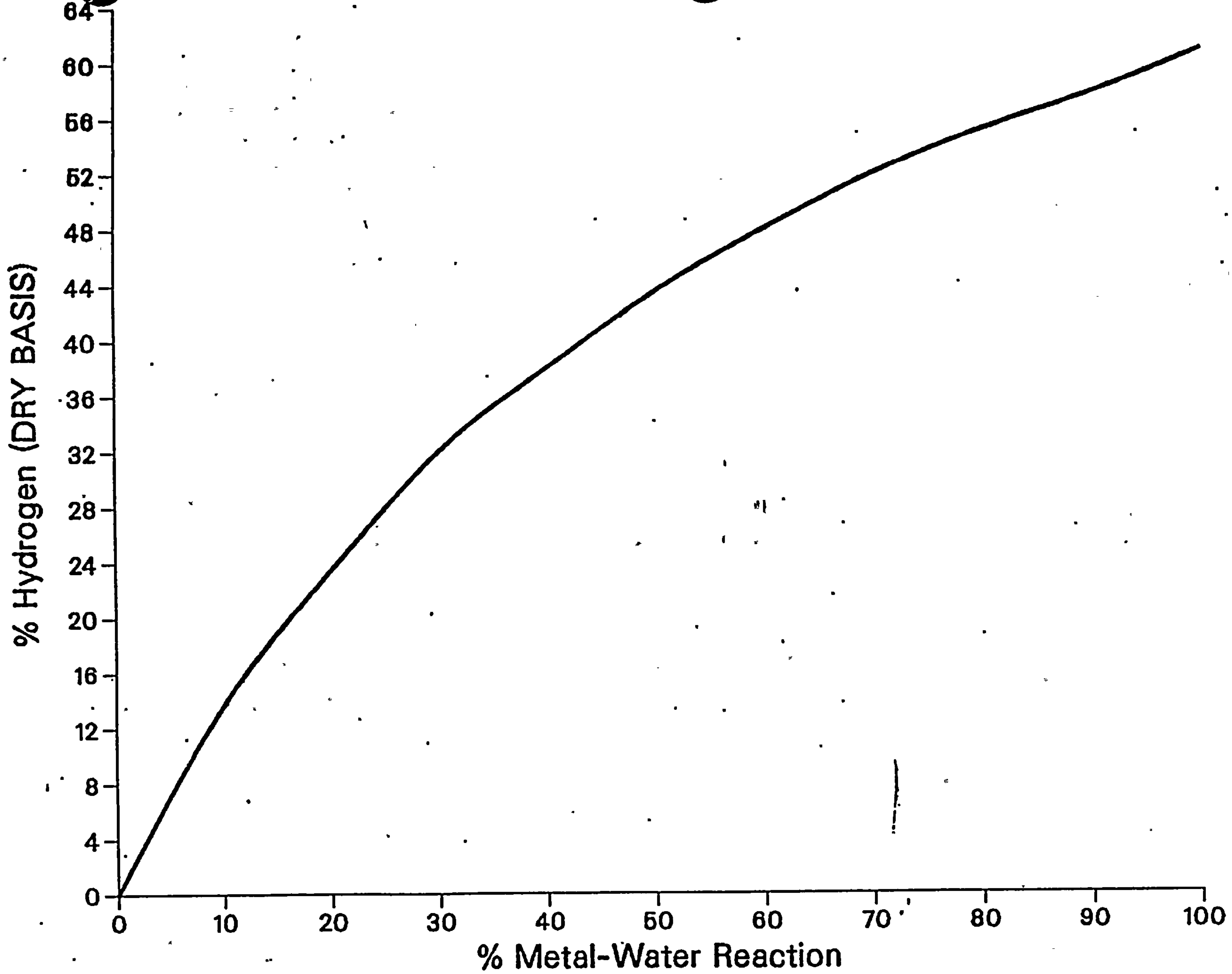
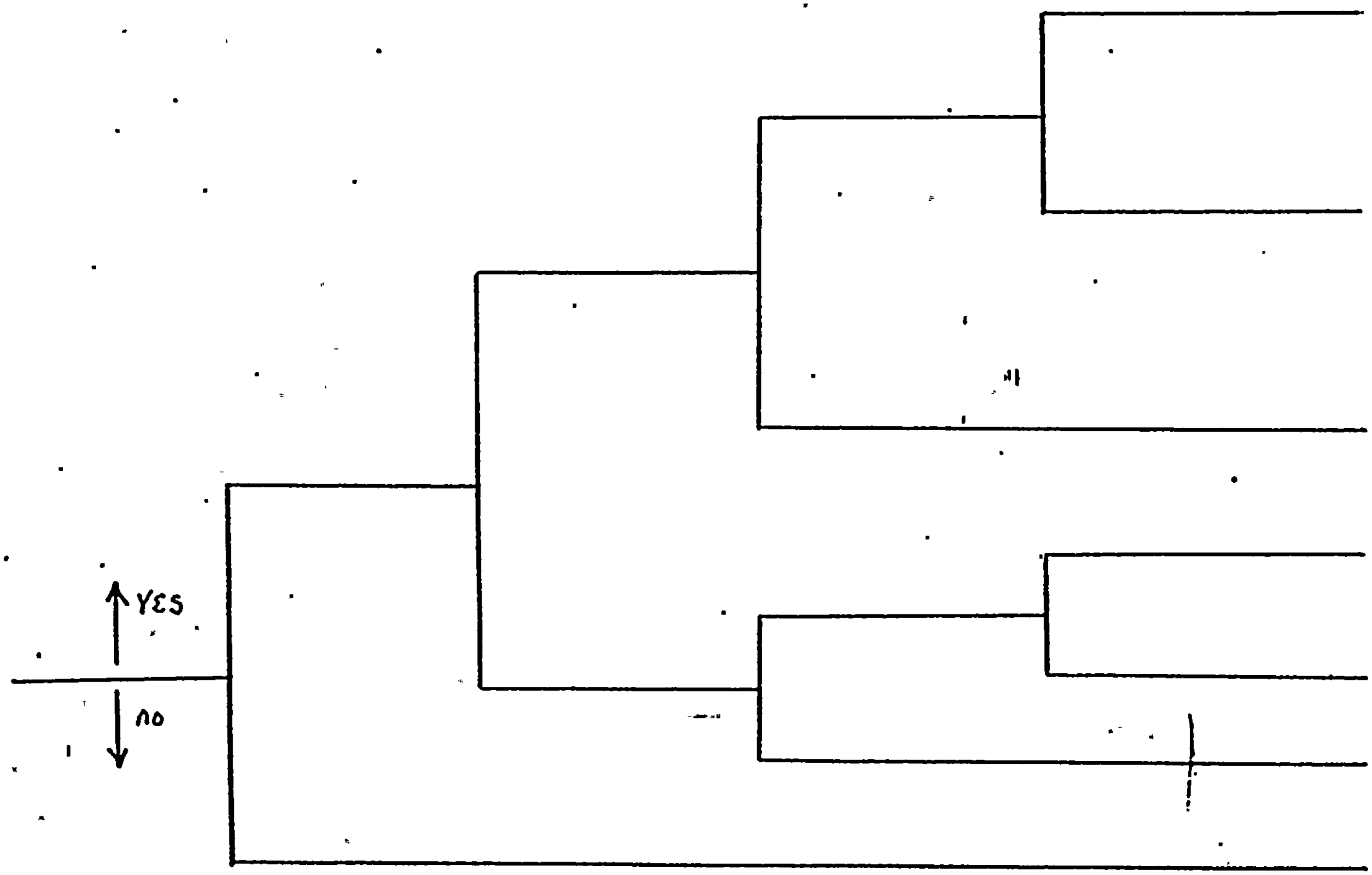


Figure 6 — Containment Hydrogen Concentration As A Function Of Metal-Water Reaction

Core damage estimate required based on procedure	PASS samples indicate clad failure	Maximum core uncovery time exceded	Containment hydrogen concentration indicates overheat	Fission products coolant sample indicates fuel melting	NRC Core Damage Index



≤10, clad failure, fuel melt indicated. Use less volatile elements to obtain melt fraction.

≤7, clad failure, fuel overheat indicated, but no melting. Use. PASS estimate to obtain index.

≤4, clad failure, but no fuel overheat indicated. Use PASS clad failure estimate to obtain index.

8, less than 10% fuel melting.

5, less than 10% fuel fuel overheat.

2, less than 10% clad failure.

1, no damage indicated.

Figure 7 – LOGIC USED TO DETERMINE NRC CORE DAMAGE INDEX

Basic Data needed to Compute Isotope Correction Factors

1. Time the reactor was shutdown.
2. Time the isotopic specific activities are recorded.
3. Time difference between shutdown and sampling time (Item 2-Item 1)
4. Containment Pressure
5. Containment Temperature
6. Sample Pressure
7. Sample Temperature
8. Isotopic Concentrations

8. Isotopic Concentrations	I-131	Cs-137	Xe-133	Kr-85
a. Reactor water (μci/gm)				
b. Suppression pool concentration (μci/gm)				
c. Drywell air concentration (μci/cc)				
d. Wetwell air space concentration (μci/cc)				

Computation of the Corrected
Fission Product Concentrations

	I131	Cs137	Xe133	Kr85
1. Computation of the Corrected Fission Product Concentration				
a. Normalized Fission Product Specific Activity (Cwi, Cgi)				
b. Thermodynamic Correction Factor (TCF)	1.0	1.0		
c. Power History Correction Factor (FIi)				
d. Plant Parameters Correction Factor (Fw, Fg)				
e. Corrected Fission Product Concentration (Ciw, Cig)				

Computation of Inventory Correction Factor for
Intermediate and Long-Lived Fission Products

	Column #	1 Decay Constant λi (days^{-1})	2 Inventory in Reference Plant $3651x(1-e^{-1095}\lambda)$	3 $\frac{\text{Column 2}}{\text{(Total from EP-IP-048-7)}}$	4 Inventory Correction Factor FIi
	ISOTOPE				
	I-131				
	Cs-137				
	Xe-133				
	Kr-85				
Intermediate	Te-132	0.2127	3651		
Intermediate	Mo-99	0.2127	3651		
Intermediate	Ba-140	0.0542	3651		
Intermediate	La-140	0.4159	3651		
Long-Lived	Ru-103	0.0176	3651		
Long-Lived	Ce-141	0.0213	3651		
Long-Lived	Ce-144	0.0024	3388		
Long-Lived	Zr-95	0.0108	3651		

Summary of Core Damage Estimates

1. Core Damage Estimate Based on PASS samples

Isotope	Clad Failure	Fuel melt
1.1 I-131		
1.2 Cs-137		
1.3 Ce-133		
1.4 Kr-85		

1.5 PASS sample indicates clad failure or fuel melt — Yes or No

2. Maximum Core uncovery time.

2.1 Time after Reactor shutdown the core became uncovered ____

2.2 Amount of time the core was uncovered ____

2.3 Maximum core uncovery time ____

2.4 Water level data indicates the maximum core uncovery time has been exceeded — Yes or No

3. Integration of Containment Hydrogen

3.1 Mole percent of hydrogen in the containment ____

3.2 % of clad undergoing metal water reaction for the reference case, MW_{ret} ____ x.76

3.3 % of clad undergoing metal water reaction for SSES, MW ____

3.4 % of metal water reaction indicates fuel overheat — Yes or No

4. Integration of Other Fission Products

4.1 Based upon FORM EP-IP-048-5, the release of less volatile fission products indicates fuel melt has occurred. (Use the average value for the best estimate, and the lower bound and upper bound of fuel melting.) — Yes or No

	Estimate of Total Curies Released						Fission Products Available for Release							Fraction of Less Volatile Fission Products Released		
	Column #	1 Reactor Water Concentration (uci/gm)	2 Curies in the Reactor (Ci)	3 Suppression Pool Concentration (uci/gm)	4 Curies in the Suppression Pool	5 Total Curies Released (2+4)	Reference Fission Product Available for Release: 6 Best Estimate 10^5 Ci	7 Upper Limit	8 Lower Limit	9 Fission Product Adjustment Factor	Fission Products Available for Release: 10 Best Estimate	11 Upper Limit	12 Lower Limit	13 Best Estimate	14 Upper Limit	15 Lower Limit
	Isotope															
Short-Lived	Sr-91						1.5	2.3	23							
	Sr-92						12.3	2.46	24.6							
	Y-92						0.372	0.124	1.24							
	Zr-97						20.7	6.3	34.5							
Intermediate	Te-132						5.49	1.83	18.3							
	Mo-99						17.3	3.46	34.6							
	Ba-140						0.55	0.184	1.84							
	La-140						0.498	0.166	1.66							
Long-Lived	Ru-103						4.65	1.55	15.5							
	Ce-141						0.483	0.161	1.61							
	Ce-144						0.387	0.129	1.29							
	Zr-95						0.483	0.161	1.61							
	Average															

Inventory Estimate of Short Lived Isotope

Column # / Isotope	1 Average Power Fraction of Time of Trip P (MWt)	2 Time since Reactor Trip T (hr)	3 Isotope Decay Constant λ (hr^{-1})	4 T	5 $e^{-\lambda T}$	6 Correction Factor $P(e^{-\lambda T})$
Sr-91			0.0730			
Sr-92			0.2558			
Y-92			0:1958			
Zr-97			0.0410			

Operating Period (j)	Day Since Startup	Operation Time Tj (days)	Time from End of j to Shutdown Tj^0 (days)	Average Power Pj (MWt)	Computation of the Actual Fission Product Inventory											
					$P_j x(1-e^{-\lambda iTj})e^{-\lambda iTj^0}$								$P_j x(1-e^{-\lambda iTj})$			
									Intermediate				Long-Lived			
					I-131	Cs-137	Xe-133	Kr-85	Te-132	Mo-99	Ba-140	La-140	Ru-103	Ce-141	Ce-144	Zr-95

DATE ____________

TIME ____________

INITIALS ____________

NRC CORE DAMAGE INDEX

(CIRCLE ONE)

DEGREE OF DEGRADATION	MINOR (<10%)	INTERMEDIATE (10%–50%)	MAJOR (>50%)
NO FUEL DAMAGE	←	1	→
CLAD FAILURE (SEE FORM B)	2	3	4
FUEL OVERHEAT (SEE FORM C)	5	6	7
FUEL MELT (SEE FORM D)	8	9	10

CURRENT CORE DAMAGE INDEX ____________

(REFERENCE ATTACHED FORM EPIP-048-4 FOR DETAILS)