STATION 1043.1F1
REVISION 1

STATION 9918.1
ORIGINAL

MIDLAND PLANT

CHP SUPERINTENDENT

ESTIMATION OF EXTENT OF CORE DAMAGE

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INFORMATION COPY

1. Prepared and Submitted

M.J. Moe / J.B. T.T. — 4/11/83
Originator — Date

2. Recommend Approval

W.J. Beck — 6/23/83
Head of Dept — Date

3.

THIS IS/~~IS NOT~~ A Q-LISTED PROCEDURE

T.C. Bergquist — 6/24/83
Chairman PRC — Date

4. PRC Recommends Approval

T.C. Bergquist — 6/24/83
Chairman PRC — Date

Ref: PRC Meeting/Review Form 124-83 of 4-13-83

Safety Evaluation Dated 4-13-83

5. Reviewed

K.E. Martin — 6/30/83
Plant QA Supt — Date

6. Approved

Bert B. Marla — 6/30/83
Plant Manager — Date

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Identification Number STATION 9918.1

Title ESTIMATION OF EXTENT OF CORE DAMAGE

Cognizant Department Head CHP Superintendent

The following paragraphs have "(later)" within them. Those identified as "Resolve Prior to Use" shall be so resolved. Remaining items shall be incorporated in the first revision following their resolution.

[signature] 1/13/83
General Manager Date

Paragraph	Resolve Prior to Use	Paragraph	Resolve Prior to Use
3.10	Yes		
3.11	Yes		
8.2.1a	Yes		
8.2.1b	Yes		
8.2.1c	Yes		
8.2.1f	Yes		
8.2.1g	Yes		
8.2.3	Yes		
Enc 3	Yes		

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ENCLOSURES

ESTIMATION OF EXTENT OF CORE DAMAGE

1. PURPOSE

To provide methods of estimating the extent of core damage during and after an accident utilizing containment area radiation monitors or samples obtained from the Post-Accident Sampling System (PASS).

2. APPLICABILITY

2.1 Method A shall be used by the Technical Support Center Staff in the assessment of the amount of core damage during the evolution of an accident or after an accident has occurred. This method may be used without obtaining samples from PASS.

2.2 Method B is to be used under the direction of the Radiological Assessment Coordinator upon completion of the post-accident sample analysis. This method will produce a more accurate estimate of the extent of core damage and shall be the preferred method.

2.3 The Site Emergency Planning Coordinator, under the direction of the CHP Superintendent, is responsible for maintenance and review of this procedure.

2.4 This procedure is only applicable after issuance of an operating license and when the Site Emergency Plan has been implemented.

3. REFERENCES

3.1 Drawing J-3115, Instrument Location Drawing Area 1 Plan of Elevation 659-0

3.2 Drawing J-3116, Instrument Location Drawing Area 1 Plan of Elevation 659-0

3.3 Drawing J-3127, Instrument Location Drawing Reactor Building Area 2 Plan of Elevation 659-0

3.4 Drawing J-3153, Instrument Location Drawing Reactor Building Area 4 Plan of Elevation 659-0

3.5 Drawing J-3143, Instrument Location Drawing Area 3 Plan of Elevation 659-0

3.6 Drawing J-3144, Instrument Location Drawing Area 3 Plan of Elevation 659-0

3.7 FSAR, Table 15A-2

3.8 NRC Regulatory Guide 1.4

3.9 STATION 1502.1, Records Collection, Storage and Maintenance

3.10 STATION 9915.2, Post-Accident Sampling, Reactor Coolant and Containment Sump (<u>Later</u>)

3.11 STATION 9915.3, Post-Accident Sampling, Containment Atmosphere (<u>Later</u>)

3.12 STATION 9915.5, Post-Accident Samples Analyses

3.13 FSAR Chapter 11.1, Source Terms

4. GENERAL INFORMATION

4.1 The core and fuel gap source terms used in Method B are based on FSAR Chapter 11.1 calculations and assumptions. The core nuclide activity values are based on core full power operation at 2552 MWt during a 310 effective full power day equilibrium cycle.

4.2 A flow chart detailing Method B is provided in Enclosure 7.

5. PREREQUISITES/INITIAL CONDITIONS

5.1 For Method A, at least one of the two monitors for the applicable unit listed in Section 8.1.1 is required to be operational and on-scale.

5.2 For Method B, the ability to analyze post-accident samples for various radionuclides is required.

6. PRECAUTIONS

6.1 Method A

6.1.1 Depending on the time in core life and the operating history, this procedure may give slightly nonconservative results. (This procedure is based on equilibrium core activities from FSAR Table 15A-2, operation at 100% power for the duration of the fuel cycle, and NRC Regulatory Guide 1.4 assumptions for core releases.)

6.1.2 Results should be compared with the results from Method B of this procedure, if available.

6.1.3 This method is dependent upon the reactor coolant boundary being violated. Post-accident sample results are needed to estimate core damage if primary coolant leakage has not occurred.

6.1.4 This method assumes that the entire inventory of gaseous fission products are released to the primary coolant. This method assumes a LOCA has occurred and cooling is in the recirculation mode. If, in fact, the accident is a small break LOCA that is quickly contained, the amount of fission products released to containment will be less than 100%; therefore, this method will underestimate the extent of core damage.

6.2 Method B

6.2.1 Due to the short half-lives of Kr-85m (4.48 h) and Sr-92 (1.71 h), these nuclide indicators should only be measured and used in this method during the first twelve hours after reactor shutdown.

6.2.2 This procedure provides only an estimate of the extent of fuel rod failure, fuel overheating, and fuel melting. Therefore, results from this method should not be interpreted as being absolute values for core damage but as a "best estimate" based on information available immediately following an accident.

7. LIMITATIONS AND ACTIONS

7.1 None.

8. PROCEDURE

8.1 Method A: Estimating the extent of core damage from containment area radiation monitors.

8.1.1 For the applicable unit, determine the pre-accident and post-accident exposure rates (use recorders in control room if available or histogram information from Digital Radiation Monitoring System) from both of the following containment area radiation monitors:

Unit	Detector Number
1	1RE-8418
1	1RE-8440
2	2RE-8419
2	2RE-8441

Record the pre-accident and post-accident exposure rates on Enclosure 1.

8.1.2 Determine the elapsed time from the time of shutdown to the time at which the post-accident exposure rate is measured. Enter the elapsed time on Enclosure 1.

8.1.3 Determine the corrected exposure rate by subtracting the pre-accident exposure rate from the post-accident exposure

rate for the same monitor. Enter this as the corrected exposure rate on Enclosure 1.

8.1.4 Using Enclosure 2, use the elapsed time and corrected exposure rate to determine the approximate percent of fuel failure. If the fuel failure falls below 1%, multiply the corrected exposure rate by 10 and redetermine the percent of fuel failure as above. Divide this percent by 10. Enter the results on Enclosure 1.

8.1.5 Initial and date in the provided space on Enclosure 1.

8.1.6 After recording the results, submit Enclosure 1 for review and signature by the Health Physicist and the Site Emergency Director.

8.2 Method B - Estimating the extent of core damage utilizing samples obtained from PASS.

8.2.1 Determine and record the following plant parameters from data recorded at reactor shutdown and at subsequent hourly intervals up until the time of post-accident sampling. Only record the value at the time of post-accident sampling if so stated.

8.2.1.a For the applicable unit, view the Safety Parameter Display System (SPDS) display (Later), "Core Exit Fluid Temperature for Inadequate Core Cooling" diagram at the time of post-accident

sampling or as soon as possible thereafter. Determine the operation region and enter the region number in Enclosure 3 (Page 1).

8.2.1.b For the applicable unit, record in Enclosure 3 (Page 2) the core exit thermocouple average temperature values from the SPDS computer display (Later). In the case where the SPDS display is not available, obtain the current temperature reading from the Post-Accident Monitoring (PAM) panel 1/2C-31 and enter in Enclosure 3 (Page 2). Enter source of temperature value in Enclosure 3 (Page 2).

8.2.1.c For the applicable unit, record in Enclosure 3 (Page 2) the reactor coolant system pressure values from the plant computer Post-Trip display (Later). In the case where the computer display is not available, obtain the current pressure reading from the PAM panel display 1/2C-31 and enter in Enclosure 3 (Page 2). Enter source of pressure value in Enclosure 3 (Page 2).

8.2.1.d Using Enclosure 6, Figure 4, determine the operation region for the reactor core at each time increment using the average core exit thermocouple temperature and RCS pressure values

obtained in Steps 8.2.1.b and 8.2.1.c and enter in Enclosure 3 (Page 2).

8.2.1.e Using the results from Steps 8.2.1.a and 8.2.1.d determine the most severe operating condition attained between the time of reactor shutdown and post-accident sampling. The most severe operating condition corresponds to the highest numbered core operation region determined in Steps 8.2.1.a and 8.2.1.d. Enter result in Enclosure 3 (Page 1).

8.2.1.f For the applicable unit, determine the containment hydrogen volume percent from the plant computer display (Later) and enter in Enclosure 3 (Page 3). In the case where the computer is unavailable, obtain the volume percent value from the PAM panel display 1/2C-31 and enter in Enclosure 3 (Page 3). Enter source of value in Enclosure 3 (Page 3).

8.2.1.g For the applicable unit, determine the containment radioactivity levels from the high range monitors (1RE-8418, 1RE-8440 or 2RE-8419, 2RE-8441) from the RMS computer display (Later) and enter in Enclosure 3 (Page 3). In the case where the computer is unavailable, obtain the radiation level from Control Panel OC403 display

and enter in Enclosure 3 (Page 3). Enter source of value in Enclosure 3 (Page 3).

8.2.2 Perform Sampling and Analysis

8.2.2.a For the applicable unit, perform post-accident sampling as described in Reference 3.10 and 3.11, (Post-Accident Sampling, Reactor Coolant and Containment Sump) and 3.11 (Post-Accident Sampling, Containment Atmosphere).

8.2.2.b Analyze reactor coolant samples as described in Reference 3.12, Post-Accident Samples Analyses. The measured nuclide concentrations will be adjusted to take into account radioactive decay from the time of reactor shutdown. Enclosure 4 is provided to accomplish this.

8.2.2.b.1 Obtain the measured I-131 concentration (μCi/g) in the reactor coolant. Correct to time of shutdown using Enclosure 4 and enter the measured and corrected values in Enclosure 3 (Page 4).

8.2.2.b.2 Obtain the measured Cs-137 concentration (μCi/g) in the reactor coolant. Correct to time of shutdown using Enclosure 4 and enter the

measured and corrected values in Enclosure 3 (Page 4).

8.2.2.b.3 Obtain the measured La-140 concentration (μCi/g) in the reactor coolant. Correct to time of shutdown using Enclosure 4 and enter the measured corrected values in Enclosure 3 (Page 4).

8.2.2.b.4 Obtain the measured Sr-92 concentration (μCi/g) in the reactor coolant. Correct to time of shutdown using Enclosure 4 and enter the measured and corrected values in Enclosure 3 (Page 4).

8.2.2.b.5 Obtain the measured Sr-91 concentration (μCi/g) in the reactor coolant. Correct to time of shutdown using Enclosure 4 and enter the measured and corrected values in Enclosure 3 (Page 4).

8.2.2.b.6 Obtain the total concentration in the reactor coolant. This will be the sum of all radionuclides present

corrected to time of shutdown. Enter this value in Enclosure 3 (Page 4).

8.2.2.b.7 Compare the La-140, Sr-91, and Sr-92 corrected concentrations determined in Steps 8.2.2.b.3, 8.2.2.b.4 and 8.2.2.b.5 with the 50% core gap release values shown in Step 8.2.3.d.2. If any one of the corrected nuclide concentrations is equal to or greater than the 50% core gap releases values, enter "yes" in Enclosure 1 (Page 4).

8.2.2.b.8 Obtain the measured hydrogen gas concentration (cc/kg) in the reactor coolant. Enter measured value in Enclosure 3 (Page 4a).

8.2.2.c Analyze containment sump sample as described in Reference 3.12, Post-Accident Sample Analyses.

8.2.2.c.1 Obtain the measured La-140 concentration (μCi/g) in the containment sump sample. Correct to time of shutdown using Enclosure 4 and enter the measured and corrected values in Enclosure 3 (Page 5).

8.2.2.c.2 Obtain the measured Sr-92 concentration (μCi/g) in the containment sump sample. Correct to time of shutdown using Enclosure 4 and enter the measured and corrected values in Enclosure 2 (Page 5).

8.2.2.c.3 Obtain the measured Sr-91 concentration (μCi/g) in the containment sump sample. Correct to time of shutdown using Enclosure 4 and enter the measured and corrected values in Enclosure 3 (Page 5).

8.2.2.d Analyze containment atmosphere sample as described in Reference 3.12, Post-Accident Samples Analyses.

8.2.2.d.1 Obtain the measured Xe-133 concentration (μCi/cc) in the containment atmosphere. Correct to time of shutdown using Enclosure 4 and enter the measured and corrected values in Enclosure 3 (Page 5).

8.2.2.d.2 Obtain the measured Kr-85m concentration (μCi/cc) in the containment atmosphere. Correct to

time of shutdown using Enclosure 4 and enter the measured and corrected values in Enclosure 3 (Page 5).

8.2.3 Determine Core Condition

The following section contains a description of each of the four core conditions: normal operation, observable macroscopic clad damage, severe fuel overheating, and fuel melting. The following steps will be used to identify the core condition based on plant information previously measured and recorded in Steps 8.2.1 and 8.2.2. Make the appropriate entries in Enclosure 3 (Page 6) using the information in the following Steps 8.2.3.a.2, 8.2.3.b.2, 8.2.3.c.2, and 8.2.3.d.2. Identify the core conditions by the existence of at least two out of three indications present for Conditions I, III, and IV. Condition II will be identified only if all three indications are present. Make the appropriate entry at the bottom of Enclosure 3 (Page 6). If more than one is identified, enter the most severe condition.

Determine how many effective full power days of operation have occurred in the current cycle at the time of reactor shutdown from the plant computer display (<u>Later</u>). Enter effective full power days in Enclosure 5 (Page 1). Use Enclosure 5 (Page 1) with interpolation techniques to obtain core inventory correction factors and enter the

factors in Enclosure 5 (Page 1). Depending on the core condition identified at the bottom of Enclosure 3 (Page 6), estimate the percent failed fuel, percent fuel overheated, or percent fuel melted using the appropriate procedure in Steps 8.2.3.a.3, 8.2.3.b.3, 8.2.3.c.3, and 8.2.3.d.3.

8.2.3.a Condition I - Normal Operation

8.2.3.a.1 Conditions of Normal Operation:

Normal reactor operation at any power, or shutdown with no unusual conditions prior to shutdown. Adequate core cooling has been maintained. All core exit thermocouple temperatures indicate subcooled reactor coolant (operation in Region 1 of Enclosure 6, Figure 4).

8.2.3.a.2 Indications of Normal Operation:

(a) Total concentration in reactor coolant is less than 200 μCi/g (1% failed fuel).

(b) Reactor coolant and containment hydrogen levels at or below

normal operating levels (50 cc/kg, ~ 0 Vol %).

(c) All core exit thermocouple temperatures along with reactor coolant system pressure indicate operation in subcooled region of Enclosure 6, Figure 1 (Region 1).

8.2.3.a.3 Estimation of % Failed Fuel

Divide the I-131 concentration determined in Step 8.2.2.b.1 by the appropriate core inventory correction factor in Enclosure 5 (Page 1). Enter the adjusted value in Enclosure 3 (Page 7). Make an estimation of % failed fuel using Enclosure 6, Figure 2. Enter in Enclosure 3 (Page 7).

8.2.3.b Condition II - Observable Macroscopic Clad Damage

8.2.3.b.1 Conditions of Observable Macroscopic Clad Damage:

Normal reactor operation at any power or shutdown where some mechanical

clad failure is indicated. The core has adequate cooling and no significant fuel over temperature is observed. All core exit thermocouple temperatures indicate operation in subcooled Region 1 or Region 2 (superheated steam but not severe enough to cause cladding damage) as shown on Enclosure 6, Figure 1.

8.2.3.a.2 Indications of Observable Macroscopic Clad Damage

(a) Total concentration in reactor coolant is greater than 200 µCi/cc (1% failed fuel level).

(b) Reactor coolant and containment hydrogen levels at or below normal operating levels (50 cc/kg, ~ 0 Vol %).

(c) Core exit thermocouple temperatures along with reactor coolant system pressure indicate operation in Region 1 or 2 as shown on Enclosure 6, Figure 1.

8.2.3.b.3 Estimation of % Failed Fuel

Divide the I-131 concentration determined in Step 8.2.2.b.1 by the appropriate core inventory correction factor in Enclosure 5 (Page 1). Enter the adjusted value in Enclosure 3 (Page 7). Make an estimation of % failed fuel using Enclosure 6, Figure 3. Enter in Enclosure 3 (Page 7).

8.2.3.c Condition III - Severe Fuel Overheating

8.2.3.c.1 Conditions of Severe Fuel Overheating:

Abnormal shutdown conditions where it is suspected that the fuel has been at least partially uncovered for a period of time greater than a few minutes. Voiding in the core is detected by high core exit thermocouple temperatures and loss of reactor coolant saturation margin indicated by operation in Region 3 of Enclosure 6, Graph 1. Fuel clad oxidation is detected by excess

hydrogen in the containment atmosphere or in the reactor coolant; no fuel melting is suspected.

8.2.3.c.2 Indications of Severe Fuel Overheating:

(a) High range containment monitors 1RE-8418, 1RE-8440, 2RE-8419, or 2RE-8441 show post-accident radiation levels greater than 1000 R/hr during the first 10 hours after shutdown.

(b) Reactor coolant hydrogen concentration greater than 50 cc/kg or containment hydrogen levels indicative of greater than 10% clad-water interaction (.58 vol % H_2 post-LOCA steam environment or 2.47 vol % H_2 dry containment).

(c) Core exit thermocouple temperatures along with reactor coolant system pressure indicate operation in Region 3 of

Enclosure 6, Figure 1 for extended periods.

8.2.3.c.3 Estimation of % Core Overheated

Fuel overheating can occur after two different LOCA scenarios; isolatable or nonisolatable. An isolatable LOCA is when the source of primary coolant loss has been isolated early in the accident with minimal coolant spillage into the reactor building sump. A nonisolatable LOCA is when the source of primary coolant leakage has not been isolated and continual loss of primary inventory to the reactor building sump occurs. Core cooling would be reestablished in the piggy-back mode of operation: coolant injection by the HPI System and recirculation through the reactor building sump and DHR heat exchangers. Determine the % of core overheated assuming both isolatable and nonisolatable LOCA using the following:

(a) Isolatable LOCA

Divide I-131 and Cs-137 concentration determined in Steps 8.2.2.b.1 and 8.2.2.b.2 by the appropriate core inventory correction factor in Enclosure 5 (Page 1). Enter both adjusted values in Enclosure 1 (Page 7). Make an estimation of % core overheated, using Enclosure 6, Figure 4 for both isotopes. Enter both values in Enclosure 3 (Page 7). Average both values and enter in Enclosure 3 (Page 7).

(b) Nonisolatable LOCA

Divide Xe-133 and Kr-85m concentration determined in Steps 8.2.2.d.1 and 8.2.2.d.2 by the appropriate core inventory correction factor in Enclosure 5 (Page 1). Enter both adjusted values in Enclosure 3 (Page 7). Make an estimation of % core overheated, using Enclosure 6,

Figure 5, for both isotopes. Enter both values in Enclosure 3 (Page 7). Average both values and enter in Enclosure 3 (Page 7).

8.2.3.c.4 Percent Core Overheated

Determine which % core overheated value is greater from the two values determined for the isolatable and nonisolatable LOCA cases. Enter the greater of the two values in Enclosure 3 (Page 7).

8.2.3.d Condition IV - Fuel Melting

8.2.3.d.1 Conditions of Fuel Melting

Severe accident where there has been a loss of shutdown cooling and the core is uncovered for a long period of time. Core exit thermocouple temperatures indicate operation in Region 4 of Enclosure 6, Graph 1 for long periods of time. Fuel melting is suspected (ie, fuel temperature exceeds 5,000°F).

8.2.3.d.2 Indications of Fuel Melting:

(a) High range containment Monitors 1RE-8418, 1RE-8440, 2RE-8419, or 2RE-8441 show post-accident radiation levels greater than 10,000 R/hr during the first 10 hours after an accident.

(b) Core exit thermocouple temperatures along with Reactor Coolant System pressure readings indicate operation in Region 4 of Enclosure 6, Graph 1 for extended periods.

(c) Certain core fission products are detectable in the reactor coolant or containment sump samples in excess of normal operating concentrations. This indicates nuclide releases from the fuel pellet. These nuclides include La-140, Sr-92, and Sr-91.

Normal RCS Concentration (μCi/g)

(Based on 1% Failed Fuel)

La-140	2.45×10^{-3}
Sr-92	1.12×10^{-3}
Sr-91	5.95×10^{-3}

Accident RCS Concentration (μCi/g)

(Based on 50% Core Gap Release

La-140	$5.20 \times 10^{+0}$
Sr-92	6.45×10^{-2}
Sr-91	4.54×10^{-1}

8.2.3.d.3 Estimation of % Core Melted

Divide Xe-133 and Kr-85m concentrations determined in Steps 8.2.2.d.1 and 8.2.2.d.2 by the appropriate core inventory correction factor in Enclosure 5 (Page 1). Enter both adjusted values in Enclosure 3 (Page 7). Make an estimation of % core melted using Enclosure 6, Figure 6, for both isotopes. Enter both values in Enclosure 3 (Page 7). Average both

values and enter in Enclosure 3 (Page 7).

8.2.4 After recording results, submit Enclosure 3 for approval by the Site Emergency Director.

8.3 All records shall be maintained in accordance with Reference 3.9, (STATION 1502.1, Records Collection, Storage and Maintenance.)

9. CHECKOFF LISTS

9.1 None.

10. ENCLOSURES

10.1 Data Sheet for Determining Extent of Core Damage.

10.2 Corrected Exposure Rate vs Elapsed Time.

10.3 Data Sheet for Determining the Extent of Core Damage (pp 1-7).

10.4 Nuclide Parameters (p 1).

10.5 Core Inventory Correction Factors (p 1).

10.6 Graphs.

10.6.1 Core Exit Fluid Temperature for Inadequate Core Cooling (p 1).

10.6.2 Case I - Normal Operation - Expected I-131 Activity Concentration in Reactor Coolant vs Percent Failed Fuel (p 2).

10.6.3 Case II - Observable Macroscopic Clad Damage - Expected I-131 Concentration in Reactor Coolant vs Percent Failed Fuel (p 3).

10.6.4 Case III - Severe Fuel Overheating - Expected Nuclide Concentration in Reactor Coolant vs Percent Core Overheated (p 4).

10.6.5 Case III - Severe Fuel Overheating - Expected Nuclide Concentration in Containment Atmosphere vs Percent Core Overheated (p 5).

10.4.6 Case IV - Fuel Melting - Expected Nuclide Containment Concentration vs Percent Core Melted (p 6).

10.7 Flowchart (p 1).

11 DISTRIBUTION

A.3

A.5

G.1

DATA SHEET FOR DETERMINING EXTENT OF CORE DAMAGE
(Method A)

ELAPSED TIME(HRS)	MONITOR NUMBER	PRE-ACCIDENT EXPOSURE RATE(R/hr)	POST-ACCIDENT EXPOSURE RATE (R/hr)	CORRECTED EXPOSURE RATE (R/hr)	PERCENT FUEL FAILURE (%)	CALCULATED BY/DATE
	1RE-8418					
	1RE-8440					
	2RE-8419					
	2RE-8441					
	1RE-8418					
	1RE-8440					
	2RE-8419					
	2RE-8441					

Reviewed By:

______________________ Health Physicist ______ Date

______________________ Site Emergency Director ______ Date

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ENCLOSURE 2
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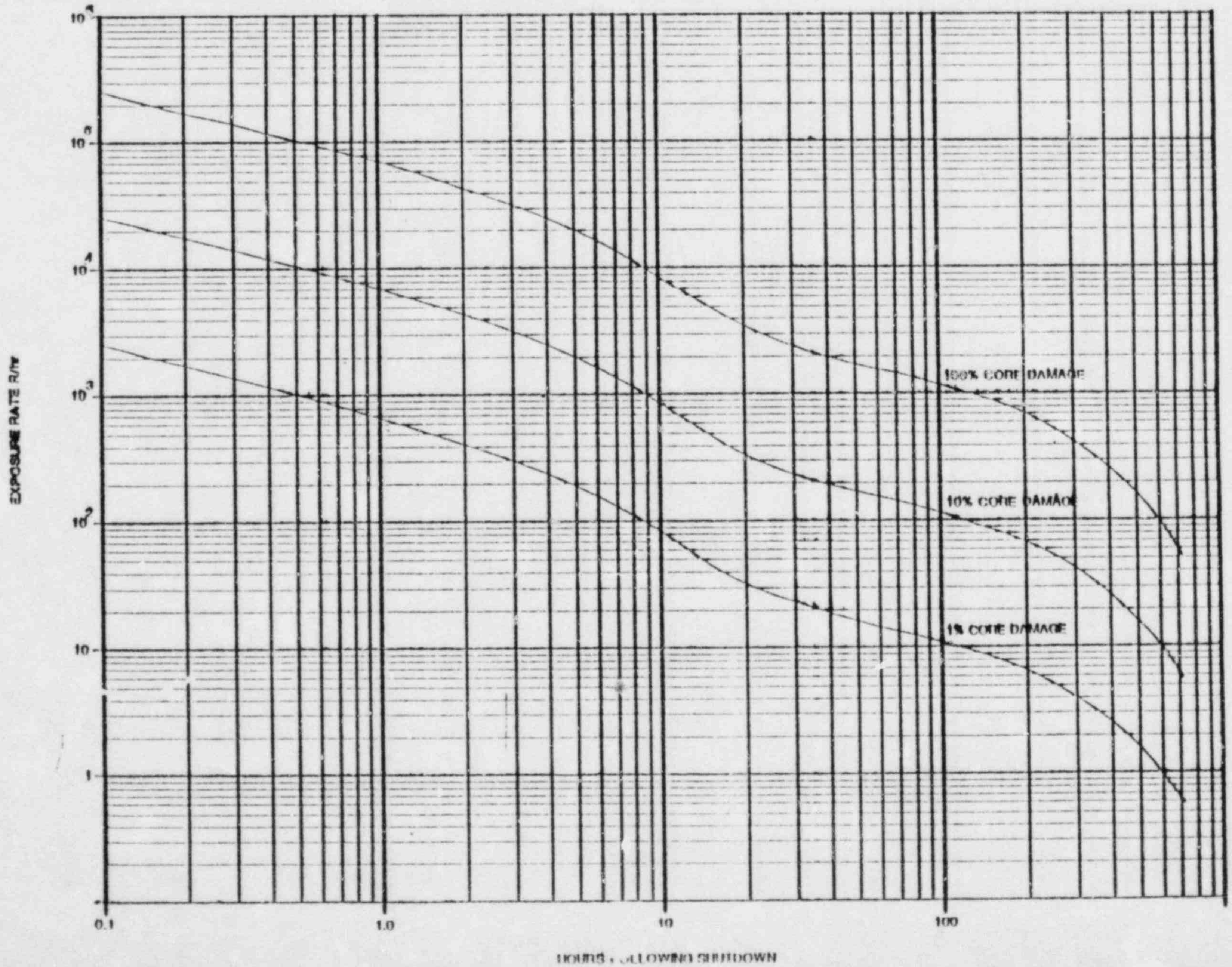
ORIGINAL
PAGE 1

EXPOSURE RATE vs ELAPSED TIME for CONTAINMENT MONITORS
1RE - 0418 1RE - 0440 2RE - 0419 2RE - 0441



DATA SHEET FOR DETERMINING THE EXTENT OF CORE DAMAGE
(Method B)

Step 8.2.1a SPDS Inadequate Core Cooling Diagram (Computer Display (Later)

Time Operating Region

RC Subcooled - Region 1

RC Superheated - Region 2

RC Superheated - Region 3

RC Superheated - Region 4

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Step 8.2.1e Final Result - Core Operating Condition

RC Subcooled - Region 1 ___

RC Superheated - Region 2 ___

RC Superheated - Region 3 ___

RC Superheated - Region 4 ___

Performed by: ______________

Date: ______________

Step 8.2.1b Core Exit Thermocouple Display
Step 8.2.1c (Computer Display (Later))
Step 8.2.1d (PAM Panel Display (1/2C31)

1 - RC Subcooled
2 - RC Superheated
3 - RC Superheated
4 - RC Superheated

Time	Avg Temp (°F)	Source of Value	RCS Pressure (psia)	Source of Value	Operating Region

Performed by: ____________________
Date: ____________________

Step 8.2.1f Containment Hydrogen Monitor (Computer Display (Later))

Time	Volume %	Source of Value

Step 8.2.1g High Range Containment Radiation Monitor

(RMS Display (Later))
OC403 Display (Later))

Time	Radiation Level (R/hr)	Source of Value

Performed by: ____________________
Date: ____________________

Step 8.2.2b Reactor Coolant Sample

Time of Sample	Date	I-131 Concentration (μCi/g) Measured/Corrected	Cs-137 Concentration (μCi/g) Measured/Corrected

Time of Sample	Date	Concentration (μCi/g) Measured/Corrected La-140	Sr-92	Sr-91	Concentration > 50% Gap Release?

Time of Sample	Date	Total Concentration (μCi/g)

Performed by: ____________________
Date: ____________________

Step 8.2.2b Reactor Coolant Sample

Time of Sample	Date	Hydrogen Concentration (cc/kg)

Performed by: ____________________
Date: ____________________

Step 8.2.2c Containment Sump Sample

Time of Sample	Date	La-140 Concentration (μCi/g) Measured/Corrected	Sr-92 Concentration (μCi/g) Measured/Corrected	Sr-91 Concentration (μCi/g) Measured/Corrected

Step 8.2.2d Containment Atmosphere Sample

Time of Sample	Date	Xe-133 Concentration (μCi/cc) Measured/Corrected	Kr-85m Concentration (μCi/cc) Measured/Corrected

Performed by: ______________
Date: ______________

CORE CONDITION
INDICATION CHECKLIST

(Check box if indication is met as described in procedure)
(Condition identified by existence of at least 2 out of 3 indications)

Step 8.2.3a2 Normal Operation - Condition I

Indication 1. ___ Total concentration in reactor coolant < 200 µCi/g (Enclosure 3, Page 4)

Indication 2. ___ H_2 levels < 50 cc/kg, ~ 0 Vol % (Enclosure 3, Page 3, 4a)

Indication 3. ___ Core T/C show subcooled RC (Region 1) (Enclosure 3, Page 1)

Step 8.2.3b2 Observable Macroscopic Clad Damage - Condition II

Indication 1. ___ Total concentration in reactor coolant > 200 µCi/g (Enclosure 3, Page 4)

Indication 2. ___ H_2 levels < 50 cc/kg, ~ 0 Vol % (Enclosure 3, Page 3, 4a)

Indication 3. ___ Core T/C show subcooled RC (Region 1) (Enclosure 3, Page 1) or superheated RC (Region 2)

Step 8.2.3c2 Severe Fuel Overheating - Condition III

Indication 1. ___ High range containment monitor > 10^3 R/hr and < 10^4 R/hr (Enclosure 3, Page 3)

Indication 2. ___ H_2 levels > 50 cc/kg or > .58 Vol % steam environment (or 2.47 Vol % dry environment) (Enclosure 3, Page 3, 4a)

Indication 3. ___ Core T/C show superheated RC (Region 3) (Enclosure 3, Page 1)

Step 8.2.3d2 Fuel Melting - Condition IV

Indication 1. ___ High range containment monitor > 10^4 R/hr (Enclosure 3, Page 3)

Indication 2. ___ Core T/C show superheated RC (Region 4) (Enclosure 3, Page 1)

Indication 3. ___ La-140, Sr-92 or Sr-91 RCS concentrations greater than 50% core gap release values (Enclosure 3, Page 4)

Core Condition

_____ I. Normal Operation
_____ II. Observable Macroscopic Clad Damage
_____ III. Severe Fuel Overheating
_____ IV. Fuel Melting

Performed by: ____________________
Date: ____________________

FINAL RESULTS

Step 8.2.3a3 Normal Operation - Case I

Adjusted I-131 Activity (μCi/g) ________

% Failed Fuel (From Figure 2) ________

Step 8.2.3b3 Macroscopic Clad Damage - Case II

Adjusted I-131 Activity (μCi/g) ________

% Failed Fuel (From Figure 3) ________

Step 8.2.3c3 Severe Fuel Overheating - Case III

Isolatable LOCA

Adjusted I-131 Activity (μCi/g) ________

Adjusted Cs-137 Activity (μCi/g) ________

% Core Overheated (From Figure 4) ____/____

% Core Overheated (Average) ________

Nonisolatable LOCA

Adjusted Xe-133 Activity (μCi/cc) ________

Adjusted Kr-85m Activity (μCi/cc) ________

% Core Overheated (From Figure 5) ____/____

% Core Overheated (Average) ________

Step 8.2.3c4 % Core Overheated ________

Step 8.2.3d3 Fuel Melting - Case IV

Adjusted Xe-133 Activity (μCi/cc) ________

Adjusted Kr-85m Activity (μCi/cc) ________

% Core Melted (From Figure 6) ____/____

% Core Melted (Average) ________

Performed by: ________
Date: ________
Reviewed by: ________
Engineering Coordinator
Approved by: ________
(SED)

NUCLIDE PARAMETERS

	Half Life (Hour)	Decay Constant ($Hour^{-1}$)
I-131	192.98	3.592×10^{-3}
Cs-137	2.64×10^{5}	2.626×10^{-6}
La-140	40.23	1.723×10^{-2}
Sr-91	9.52	7.281×10^{-2}
Sr-92	2.71	2.558×20^{-1}
Xe-133	126	5.501×10^{-3}
Kr-85m	4.48	1.547×10^{-1}

Activity Correction Formula

$No = Ne^{\lambda t}$

where

No = nuclide activity concentration at reactor shutdown

N = measured nuclide activity concentration at time of sampling

$T_{1/2}$ = half life (hr)

$\lambda = \ln 2/T_{1/2}$ = decay constant (hr^{-1})

t = time of sampling (in hours) after reactor shutdown

CORE INVENTORY CORRECTION FACTORS

Nuclide	Effective Full Power Days of Operation									
	4	30	60	90	120	150	180	210	280	310
Kr-85m	1.0	-(1)	-	-	-	-	-	-	-	-
Xe-133	.38	.98	1.0	-	-	-	-	-	-	-
Sr-91	1.0	-	-	-	-	-	-	-	-	-
Sr-92	1.0	-	-	-	-	-	-	-	-	-
I-131	.42	.94	.99	1.0	-	-	-	-	-	-
La-140	.41	.85	.97	.99	1.0	-	-	-	-	-
Cs-137	Use equation below									

(1) "-" = 1.0

Cs-137 Correction Factor

$F = .0016\ t + .50$

where

F = Cs-137 correction factor

t = effective full power days

Correction Factor

Kr-85m ________
Xe-133 ________
Sr-91 ________
Sr-92 ________
I-131 ________
La-140 ________
Cs-137 ________

EFPD ________

INFORMATION COPY

Performed by: ________________
Date: ________________

FIGURE 1
Core Exit Fluid Temperature For Inadequate Core Cooling

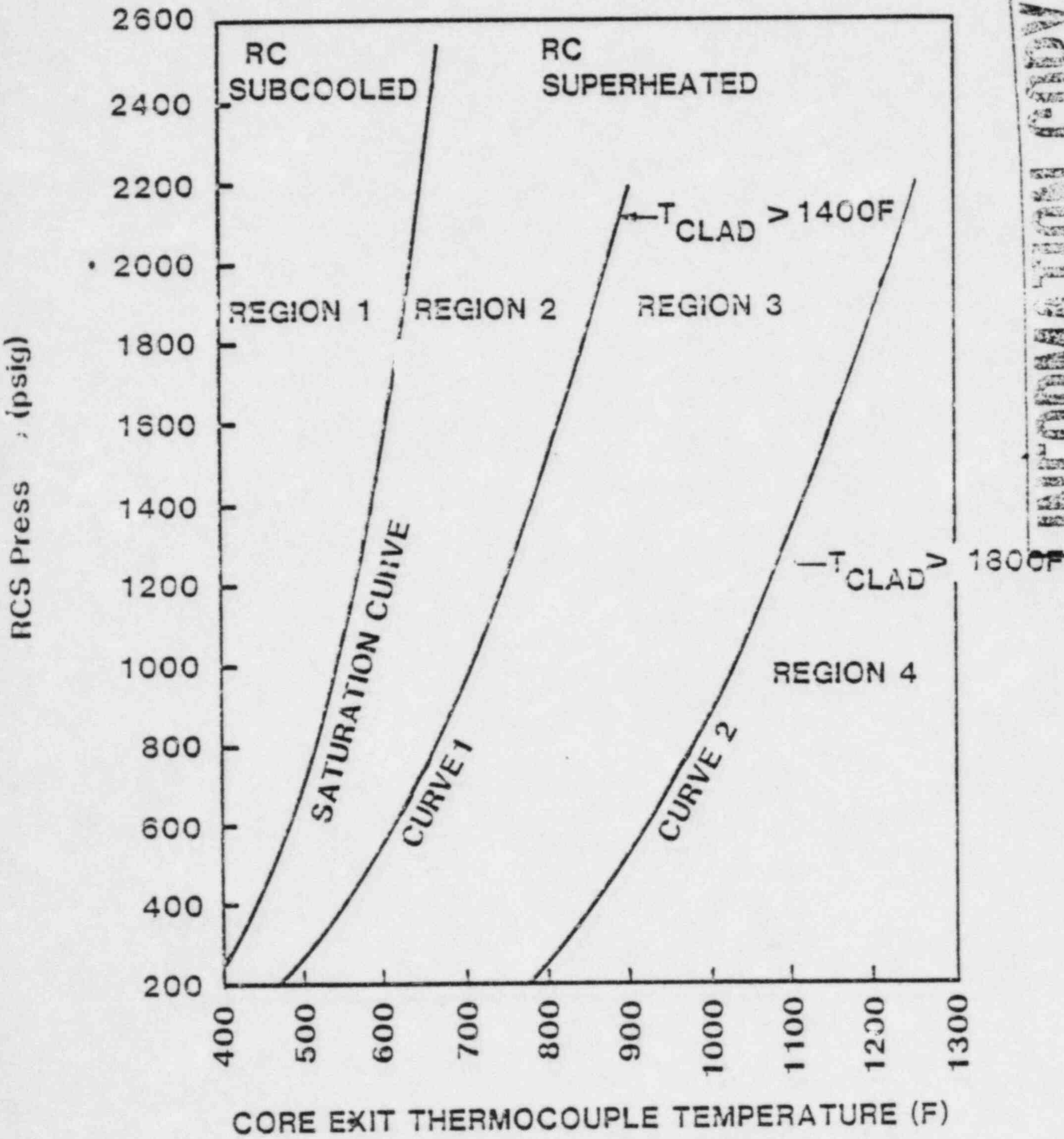


FIGURE 2

ENCLOSURE 6
STATION 9918.1
ORIGINAL
PAGE 2

CASE I - NORMAL OPERATION

EXPECTED I-131 ACTIVITY CONCENTRATION in REACTOR COOLANT

vs

PERCENT FAILED FUEL

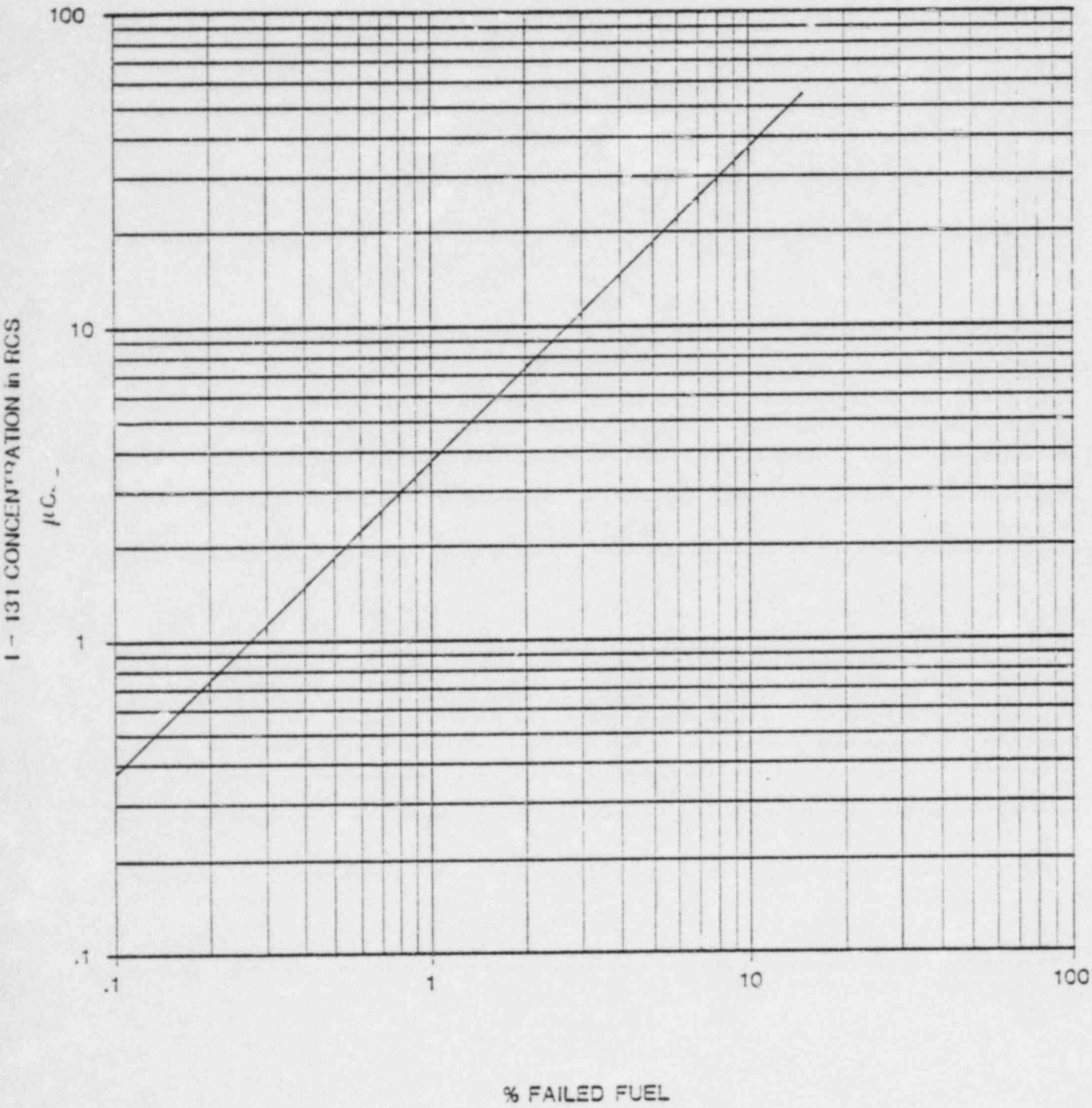


FIGURE 3

CASE II - OBSERVABLE MACROSCOPIC CLAD DAMAGE

EXPECTED I-131 CONCENTRATION in REACTOR COOLANT
vs
PERCENT FAILED FUEL

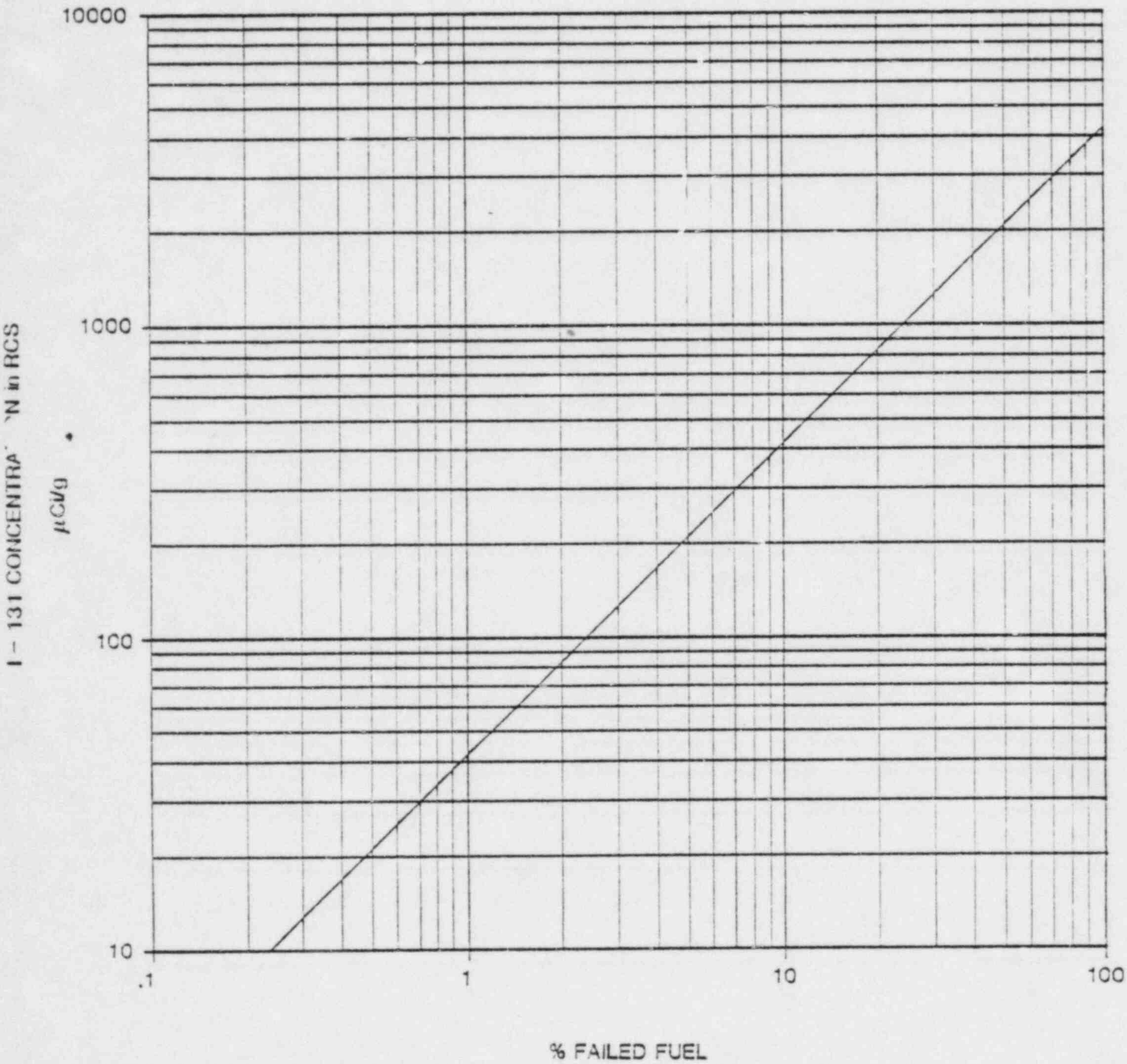


FIGURE 3

CASE II - OBSERVABLE MACROSCOPIC CLAD DAMAGE

EXPECTED I-131 CONCENTRATION in REACTOR COOLANT vs PERCENT FAILED FUEL

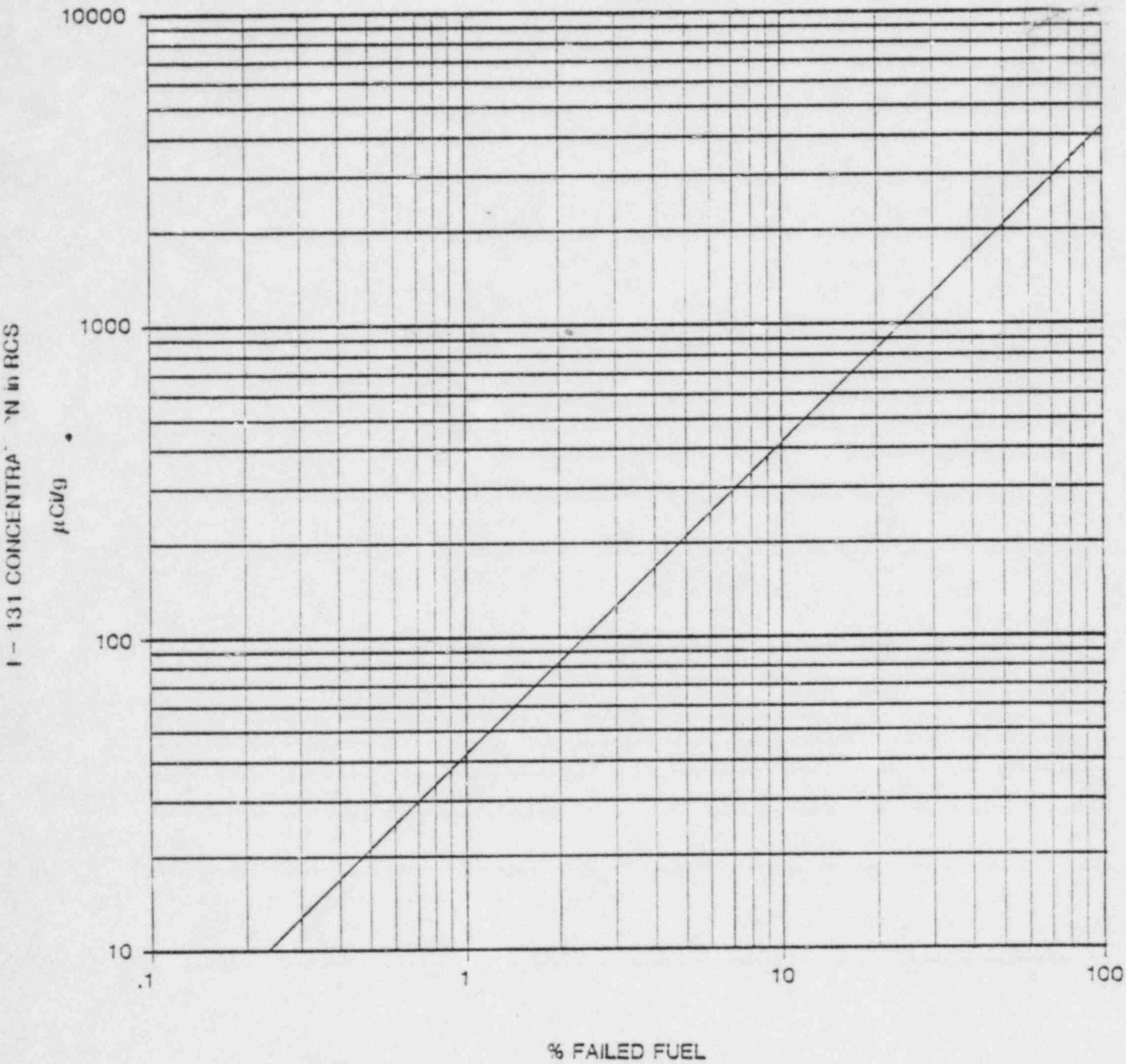


FIGURE 4

CASE III - SEVERE FUEL OVERHEATING

EXPECTED NUCLIDE CONCENTRATION in REACTOR COOLANT vs PERCENT CORE OVERHEATED

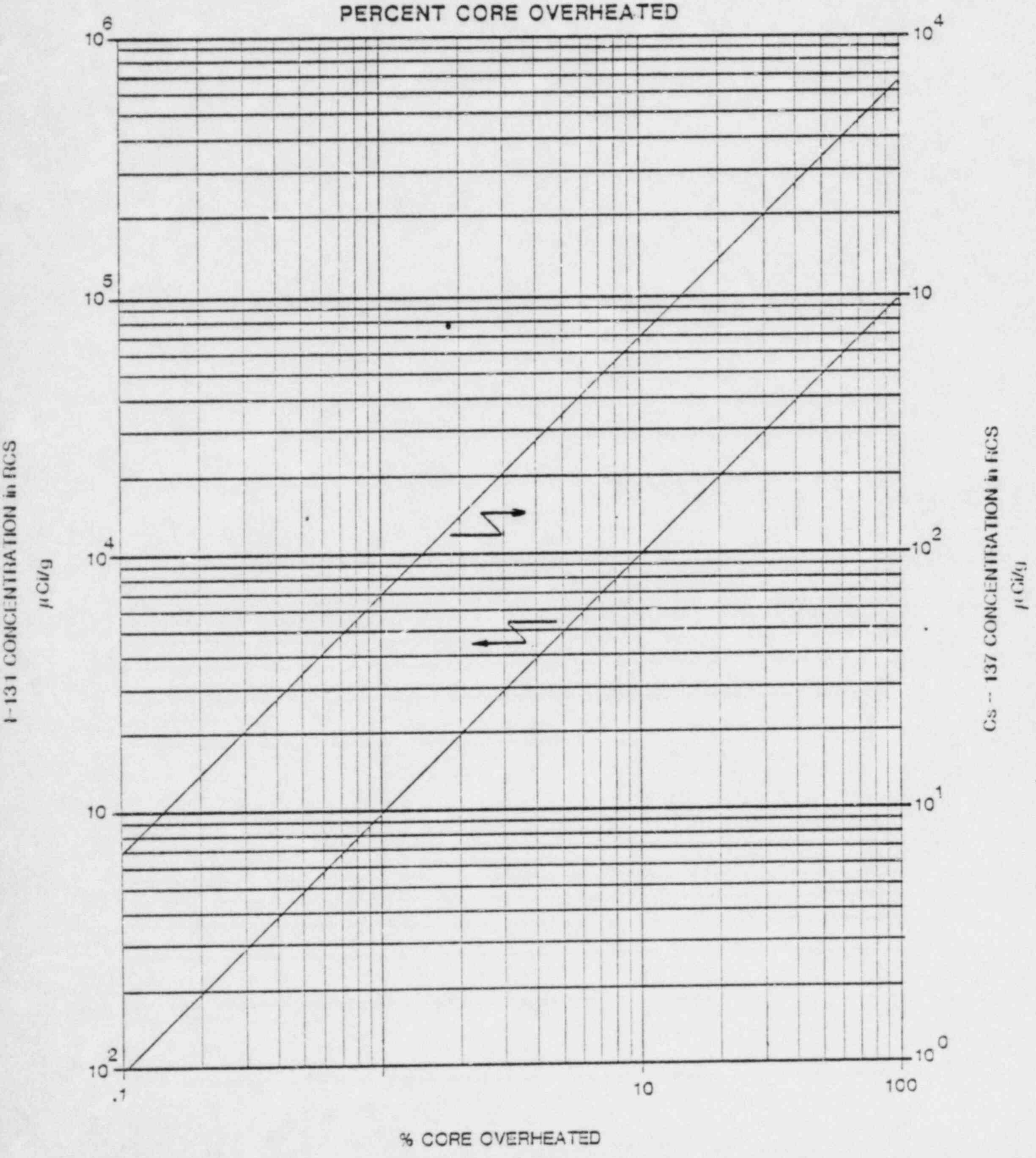


FIGURE 5

CASE III - SEVERE FUEL OVERHEATING

EXPECTED NUCLIDE CONCENTRATION in CONTAINMENT ATMOSPHERE
vs
PERCENT CORE OVERHEATED

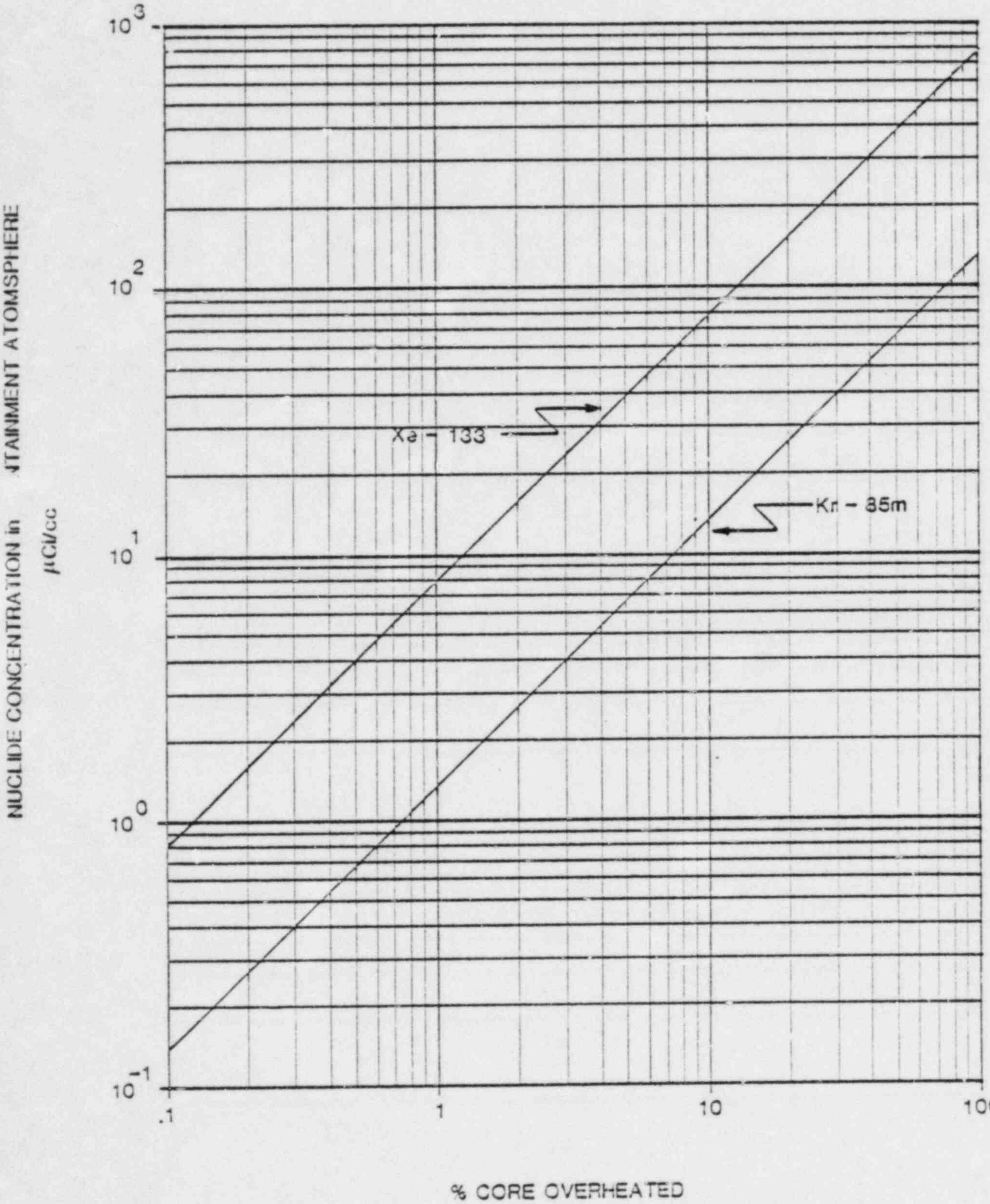
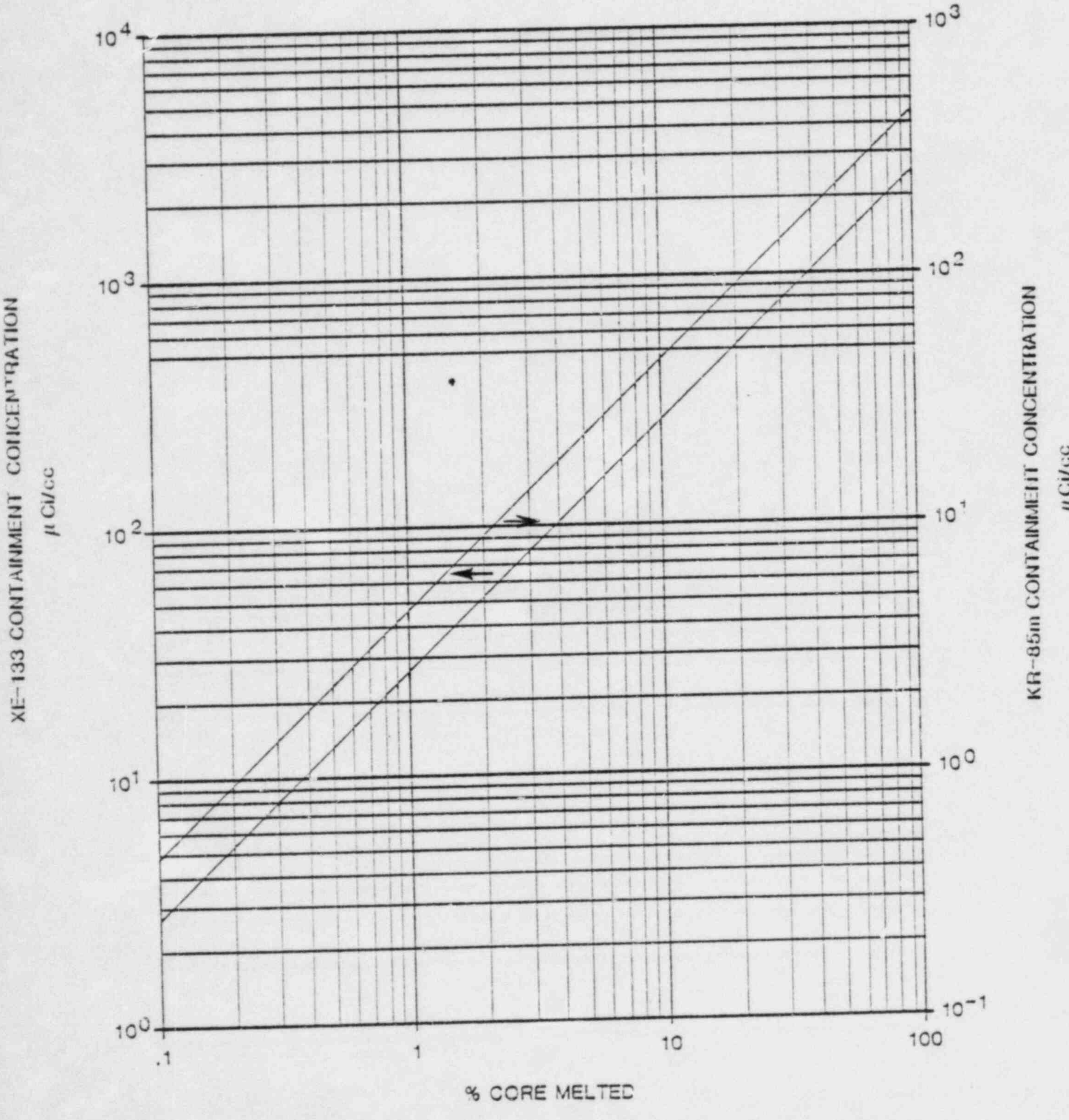


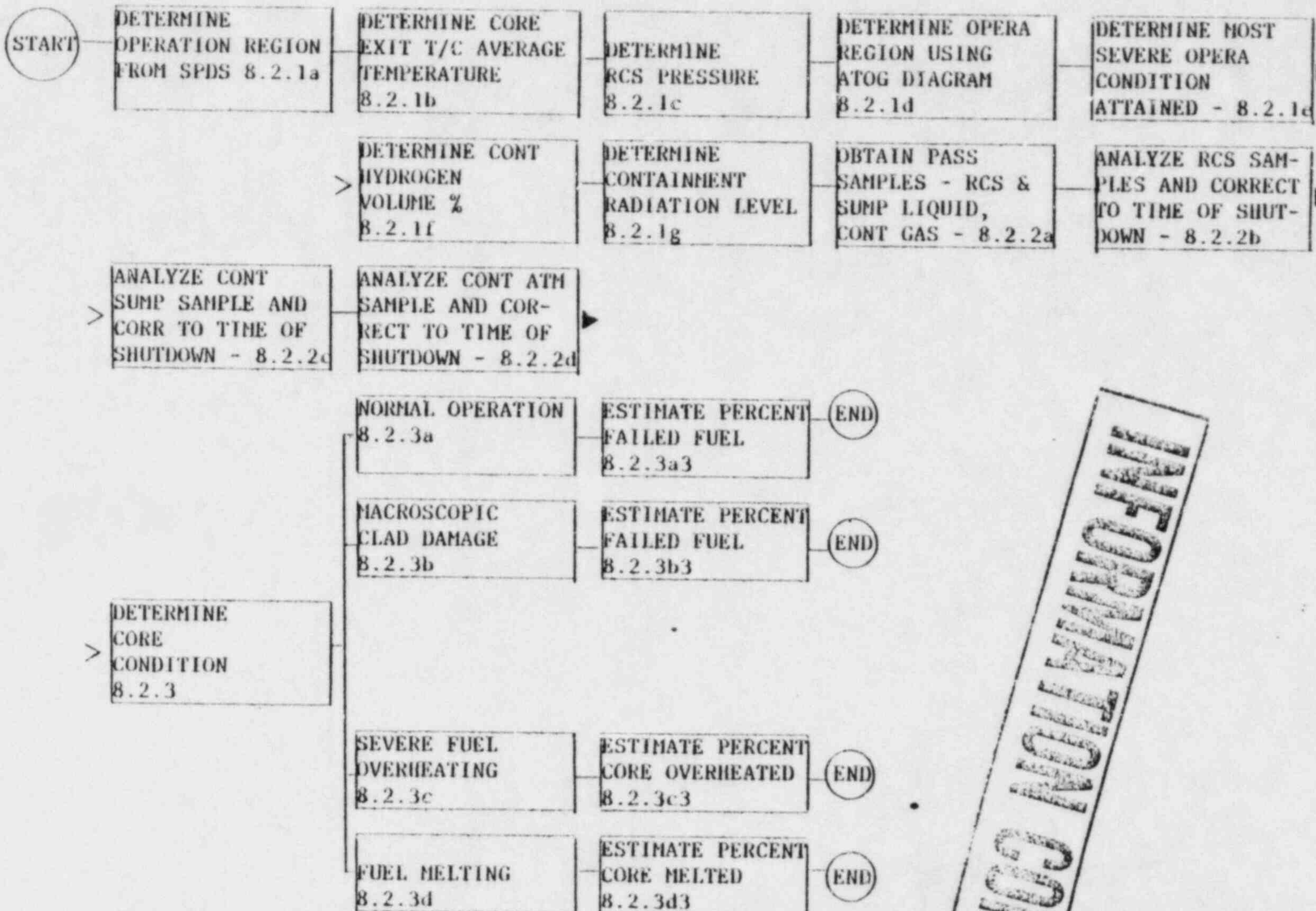
FIGURE 6

CASE IV - FUEL MELTING

EXPECTED NUCLIDE CONTAINMENT CONCENTRATION vs PERCENT CORE MELTED



FLOWCH __ METHOD B



ENCLOSURE 7
STATION 9918.1
ORIGINAL

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